FENOC

FirstEnergy Nuclear Operating Company

Thomas S. Cosgrove
Director Maintenance

724-682-5203

February 9, 2007
L-07-017

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

Subject: **Beaver Valley Power Station, Unit Nos. 1 and 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Requests Nos. 334 and 205

On September 6, 2005, via letter L-05-146 (Reference 1), FirstEnergy Nuclear Operating Company (FENOC) submitted a response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" (Reference 2). This letter contained commitments to replace the containment sump screens and to submit License Amendment Requests (LAR) that would modify the start signal for the Recirculation Spray System (RSS) pumps at both Beaver Valley Power Station (BVPS) units.

Pursuant to 10 CFR 50.90, FENOC requests amendments to the above licenses in the form of changes to the BVPS Technical Specifications. These LARs propose Technical Specification changes that will provide consistency with a proposed change to the RSS pump start signal due to a modification to the containment sump screens. The containment sump screen modifications are being made to support BVPS resolution of Nuclear Regulatory Commission (NRC) Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance" (Reference 3). Additionally, NRC approval of a change to the calculation methodology used to determine aerosol removal coefficients for use in dose consequence analyses is being requested by these LARs.

The Enclosure contains the FENOC evaluation of the proposed changes. The Enclosure contains markups of the Technical Specifications, the Technical Specification Bases and the Licensing Requirements Manual pages. The Regulatory Commitments are contained in Attachment F of the Enclosure.

The containment sump modifications will be completed during the Fall 2007 (1R18) refueling outage at Unit 1 and the Spring 2008 (2R13) refueling outage at Unit 2. Since the sump modifications will be completed during different refueling outages, the unit

A116

specific amendments will be implemented on a staggered basis. Therefore, FENOC requests approval of the proposed amendments by September 14, 2007, to support the Unit 1 Fall 2007 refueling outage. Once approved, the amendments shall be implemented as follows. The Unit 1 amendment will be implemented prior to the first entry into Mode 4 coming out of the Unit 1 Fall 2007 (1R18) refueling outage. The Unit 2 amendment will be implemented prior to the first entry into Mode 4 coming out of the Unit 2 Spring 2008 (2R13) refueling outage.

These changes have been reviewed by the Beaver Valley Power Station review committees. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazards consideration.

If there are any questions or if additional information is required, please contact Mr. Henry L. Hegrat, Supervisor - FENOC Fleet Licensing, at (330) 315-6944.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 9, 2007.

Sincerely,

Thomas S. Cosgrove

References:

1. FENOC letter L-05-146, "Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 6, 2005.
2. Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," September 13, 2004.
3. NRC Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance."

Enclosure:

FENOC Evaluation of the Proposed Changes

c: Ms. N. S. Morgan, NRR Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. J. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ENCLOSURE
FENOC Evaluation of the Proposed Changes

Beaver Valley Power Station
License Amendment Requests
334 (Unit 1) and 205 (Unit 2)

Subject: Recirculation Spray System Pump Start Signal

Table of Contents

Attachments

1.0 DESCRIPTION

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

The Beaver Valley Power Station Technical Specifications are being revised to be consistent with a proposed change to the Recirculation Spray System (RSS) pump start signal due to a modification to the containment sump screens. The first Technical Specification change consists of starting the RSS pumps on a coincident Containment Pressure High-High/Refueling Water Storage Tank (RWST) Level Low Engineered Safety Feature Actuation System (ESFAS) signal instead of a signal from a timer. The existing timer imposes a fixed time delay on the pump start following a Containment Pressure High-High signal. The proposed change to the pump start signal results in the pump start being driven by plant conditions. The resultant extended pump start assures that plant conditions are such that the pumps are started when needed and that sufficient water is present in the containment sump. The second Technical Specification change consists of modifying the nomenclature for the ESFAS signal used for automatic switchover to Emergency Core Cooling System (ECCS) suction from the containment sump. The third Technical Specification change consists of revising the containment sump 18 month visual inspection Surveillance Requirement. The fourth Technical Specification change consists of raising the upper limit on containment average air temperature to reflect a revised containment analysis. The fifth Technical Specification change consists of lowering peak containment pressure, P_a, to reflect a revised containment analysis.

Additionally, Nuclear Regulatory Commission (NRC) approval of a change to the calculation methodology used to determine aerosol removal coefficients for use in dose consequence analyses is requested. The methodology change consists of using MAAP-DBA instead of LOCTIC to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code.

All of these changes are being made to support BVPS resolution of NRC Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance" (Reference 1).

2.0 PROPOSED CHANGES

On February 25, 2005, FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) Nos. 296 (Unit 1) and 169 (Unit 2) to convert the current Beaver Valley Power Station (BVPS) Technical Specifications to the Improved Standard Technical Specifications, NUREG-1431, "Standard Technical Specification - Westinghouse Plants." The February 25, 2005 submittal (Reference 2) requested an amendment approval date of August 2006 and an implementation date of at least 150 days. Since implementation of the converted BVPS Technical Specifications is expected prior to approval of this LAR, proposed changes to the BVPS Technical Specifications are submitted in the Improved Technical Specification (ITS) format. The following discussion of the proposed changes refer to the ITS format of the BVPS Technical Specifications and Technical Specification Bases.

The proposed Technical Specification (TS) changes, which are submitted for NRC review and approval, are provided in Attachment A[1] for BVPS Units 1 and 2. The changes proposed to the Technical Specification Bases are provided in Attachment B for Units 1 and 2. With the conversion of the BVPS TS to the ITS format, both units are addressed by a single TS and TS Bases. 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 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 description of the containment and net positive suction head analyses conducted to support this LAR. Attachment E provides a description of the revised dose consequence analyses also conducted to support this LAR. Attachment F provides a list of commitments associated with this LAR.

---

[1] 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, Technical Specification 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.

CHANGE	TS	TITLE
1	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation, Table 3.3.2-1 Function 2, Containment Spray
2	3.3.2	Engineered Safety Feature Actuation System (ESFAS) Instrumentation, Table 3.3.2-1 Function 7, Automatic Switchover to Containment Sump
3	3.5.2	ECCS-Operating
4	3.6.5	Containment Air Temperature
5	5.5.12	Containment Leakage Rate Testing Program

The following provides a description of, and basis for, the proposed changes. The technical justification for the changes is provided in Section 4.0.

<u>Change No. 1</u>

Technical Specification 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, Table 3.3.2-1.

Function 2, Containment Spray, of Table 3.3.2-1 is proposed to be re-titled and modified to address two systems. The title of Function 2 is proposed to be changed to Containment Spray Systems. The existing function is proposed to be designated as Function 2.a and titled Quench Spray. The items applicable to existing Function 2 are proposed to be designated as applicable to Function 2.a, Quench Spray. A new Function, and its associated requirements (Modes, Required Channels, Conditions, Surveillance Requirements and Allowable Values), are proposed to be added and labeled as Function 2.b, Recirculation Spray. These changes are shown in Attachment A. The TS Bases and LRM are also revised to reflect these changes.

Basis for Change No. 1

The current version of Function 2 in Table 3.3.2-1 does not distinguish between the quench and recirculation spray systems. The proposed change will permit assigning the applicable requirements to the quench and recirculation spray systems individually. The requirements for the Function 2.a, Quench Spray System, are not changed by this submittal. New requirements are provided for Function 2.b, Recirculation Spray System, reflecting the change to the recirculation spray pump start signal. With the current design the recirculation spray pumps are started following a fixed time delay controlled by a timer after receipt of a Containment Pressure High-High signal. With the proposed change the recirculation spray pumps will start on a coincident Containment Pressure High-High/RWST Level Low signal. The change to the pump start signal will ensure that sufficient water is available for proper pump operation and that containment conditions are such that recirculation spray pump operation is required. The RWST Level Low Allowable Value has both upper and lower limits. The lower limit is selected to ensure that containment temperatures remain within safety analysis limits and that adequate NPSH is available to the low head safety injection (LHSI) pumps. Starting the RSS pumps at a lower level will result in a higher containment sump water temperature at switchover which decreases the LHSI pump available NPSH, but still results in adequate available NPSH. The upper limit ensures adequate NPSH to the recirculation spray pumps.

Change No. 2

Technical Specification 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, Table 3.3.2-1.

Function 7, Automatic Switchover to Containment Sump, item 7.b of Table 3.3.2-1 is proposed to be changed to Refueling Water Storage Tank (RWST) Level Extreme Low, such that Level Extreme Low will be applicable to both units. This change is shown in Attachment A. The TS Bases and LRM are also revised to reflect this change.

Basis for Change No. 2

This change is being proposed to provide consistent signal terminology between the units and to avoid confusion with the RSS pump start signal which uses the term RWST Level Low. The change is a nomenclature change only. The setpoint is not changed. With this change both units will automatically switchover ECCS suction to the containment sump on an RWST Level Extreme Low signal.

Change No. 3

Technical Specification 3.5.2, ECCS-Operating.

The proposed change modifies the requirements of Surveillance Requirement (SR) 3.5.2.7 by removal of the word "train", changing the terminology of "trash racks and screens" to "strainers" and adding the wording "accessible regions" to describe the extent of the visual inspection. The change is shown in Attachment A. The TS Bases is also revised to reflect this change. There is no change to the LRM required for this change.

Basis for Change No. 3

The proposed change will provide terminology and scope that is consistent with the modified containment sump suction inlets being installed to address Generic Safety Issue 191 (Reference 1). This change is being proposed to reflect the physical characteristics of the new containment sump strainers. The removal of the word "train" is considered to be a clarification of the current SR because the sump strainer is a combined header for both ECCS trains; hence, use of the word "train" is not needed. The terminology change from "trash racks and screens" to "strainers" provides a more appropriate description of the new configuration that strain through perforated stainless steel plates. The restriction to only "accessible regions" is due to the complexity, size and location of the new containment sump strainers.

Although the configurations of the existing trash racks and screens and the replacement sump strainer assemblies are different, they serve the same fundamental purpose of passively removing debris from the sump's suction supply to the supported system pumps. The descriptive terminology of trash racks and screens is not descriptive of the new sump strainers. The proposed replacement of "trash racks and screens" with "strainers" is a descriptive change and the revised

SR will continue to ensure the containment sump strainers are not restricted by debris and show no evidence of structural distress or abnormal corrosion. With the proposed revision there is no change to: (1) the requirement to verify that, by visual inspection, the ECCS containment sump suction inlet is not restricted by debris and the sump inlet strainers show no evidence of structural stress or abnormal corrosion or (2) the surveillance test interval of 18 months.

Change No. 4

Technical Specification 3.6.5, Containment Air Temperature.

The upper limit on containment average air temperature is proposed to be changed from 105°F to 108°F. This change is shown in Attachment A. The TS Bases is also revised to reflect this change. There is no change to the LRM required for this change.

Basis for Change No. 4

This change incorporates the revised containment analysis upper limit on containment average air temperature. Details of the analysis are provided in Section 4.0 and Attachment D.

Change No. 5

Technical Specification 5.5.12, Containment Leakage Rate Testing Program.

This change incorporates the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, from the revised containment analyses. For Unit 1, P_a is changed from 43.3 to 43.1 psig. For Unit 2, P_a is changed from 44.9 to 44.8 psig. This change is shown in Attachment A. There is no TS Bases for this Technical Specification; however the value of P_a will be changed in the Bases where appropriate. There is no change to the LRM required for this change.

Basis for Change No. 5

This change incorporates values from the revised containment analysis. Details of the analysis are provided in Section 4.0 and Attachment D.

3.0 BACKGROUND

On September 6, 2005, via letter L-05-146 (Reference 3), FENOC submitted a response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" (Reference 4). This letter contained commitments to replace the containment sump screens and to submit a LAR that would modify the start signal for the recirculation spray pumps at both BVPS units. The due date for the submittal of the LAR was modified by FENOC letters L-06-020, dated April 3, 2006 and L-06-145, dated September 29, 2006 (References 5 and 6).

During the Fall 2006 refueling outage (2R12) the Unit 2 containment sump screens underwent an interim modification that is consistent with the current Unit 2 design and licensing basis. With the interim modification the Unit 2 RSS pumps will continue to be started on a signal from a timer. The interim modification will remain in effect until a final containment sump modification has been completed at Unit 2 which is scheduled for the Spring 2008 (2R13) refueling outage. There is no interim modification planned for Unit 1. The containment sump modification to alleviate the concerns of GSI-191 and Generic Letter 2004-02 will be completed during the Fall 2007 (1R18) refueling outage at Unit 1 and the Spring 2008 (2R13) refueling outage at Unit 2. The final modifications of the containment sump for both units require approval of this LAR since the RSS pumps will then be started by an ESFAS signal. Since the final sump modifications will be completed during different refueling outages, these amendments will be implemented on a staggered basis. The Unit 1 amendment will be implemented prior to the first entry into Mode 4 coming out of the Unit 1 Fall 2007 (1R18) refueling outage. The Unit 2 amendment will be implemented prior to the first entry into Mode 4 coming out of the Unit 2 Spring 2008 (2R13) refueling outage.

The modification to replace the containment sump screens with strainers is addressed by the BVPS Engineering Change Process that requires assessment in accordance with 10 CFR 50.59. NRC approval of the design modification is not required; however a brief description is provided for each unit. The commitments to modify the start signal for the recirculation spray pumps for both units are the subject of this submittal since these changes result in changes to the BVPS Technical Specifications. This submittal satisfies the commitment made in FENOC letter L-05-146 (Reference 3) to submit a LAR to change the RSS pump start signal.

The modification involves replacing the existing screen hardware with new fabricated strainer assemblies. The strainer design was chosen based on the largest available sump strainer area that would fit within the bounds of the existing sump area and to be compatible with the anticipated water level. The new sump strainer is designed to reduce both head loss and the ingestion of debris, which could affect safety-related downstream components. The sump strainers are sized to preclude the passage of debris large enough to cause loss-of-function of any downstream components.

System Description

The containment depressurization system is composed of two systems, quench and recirculation spray. These systems are designed to provide the necessary cooling and depressurization of the containment following a design basis loss of coolant accident (LOCA). Operating together these systems cool and depressurize the containment to less then 1/2 of the peak containment pressure within 24 hours following the design basis accident, assuming the operation of at least minimum engineered safety features. Only the start signal of recirculation spray system pumps is affected by the changes being proposed in this LAR.

The BVPS Emergency Core Cooling System (ECCS) design includes several sets of pumps that reduce containment temperature and pressure and remove core heat following an accident. Following a design basis loss of coolant accident (LOCA), reactor coolant system (RCS) pressure will drop resulting in a safety injection signal (SIS) and containment pressure will rise resulting in a Containment Pressure High-High signal that results in a containment isolation phase B (CIB) signal. Upon receipt of the SIS, the high head safety injection (HHSI) pumps and low head safety injection (LHSI) pumps are started and inject water from the refueling water storage tank (RWST) into the RCS. Upon receipt of the Containment Pressure High-High signal, the quench spray system (QSS) pumps also start drawing water from the RWST and spraying into containment via spray headers to lower containment temperature and pressure. The RSS pumps will start after a time delay of approximately 300 seconds (5 minutes) at Unit 1 and 600 seconds (10 minutes) at Unit 2 following receipt of the Containment Pressure High-High signal and draw water from the containment sump and spray into containment via spray headers to assist in lowering containment temperature and pressure. When the RWST reaches its low-low level point, a transfer to the recirculation mode is initiated.

Unit 1

At Unit 1, the LHSI pumps automatically realign to take suction flow from the containment sump and supply flow to the RCS and HHSI pump suctions. Two of the four RSS pumps are located inside containment and take suction directly from the containment sump. The other two RSS pumps are outside containment and take suction from the containment sump through a dedicated line.

Unit 2

At Unit 2, the LHSI pumps automatically stop on the low-low RWST level signal and the HHSI pumps are manually realigned to take suction from the discharge of one of the two RSS pumps on each train to continue core heat removal. The other RSS pump on each train continues to discharge to its spray header to continue lowering containment temperature and pressure. At this unit all four pumps are outside containment and re-circulated containment water is provided to each RSS pump through a dedicated inlet line from the containment sump.

Units 1 and 2

The QSS pumps take suction from the RWST and discharge to spray headers until they are manually stopped based on an RWST Low Level alarm. A low level alarm is generated when approximately 1 to 2 feet of water is left in the tank. The operators manually stop the pumps when this low level alarm is received. In addition, each RSS pump discharges to a dedicated RSS heat exchanger that is cooled by service water.

Because the RSS pumps are currently started after a relatively short time delay of approximately 300 seconds (5 minutes) at Unit 1 and 600 seconds (10 minutes) at Unit 2 following a Containment Pressure High-High signal, there is a limited quantity of water in the containment sump for ECCS recirculation. As a result there is little margin to pump net positive suction head (NPSH) limits. However, this amount of margin is adequate to support the current licensing basis for operability of ECCS for each of the BVPS units.

Current Licensing Basis

The current licensing basis methodology for calculating NPSH differs from Unit 1 to Unit 2 and is briefly described in the following paragraphs.

Unit 1

At Unit 1, containment overpressure is credited in calculating the available NPSH for both the RSS pumps and the LHSI pump when taking suction from the containment sump. Therefore the equation for calculating available NPSH for Unit 1 is:

$$NPSH_a = P_{cont} - P_{vap} + H_{static} - H_{loss}$$

Where:

P_{cont} = containment total pressure

P_{vap} = vapor pressure of the fluid entering the pump

H_{static} = the static height of water above the pump impeller

H_{loss} = the friction loss in the suction piping including strainer loss

At Unit 1, a portion of the flow from the QSS is injected directly into the RSS pump suction lines to sub-cool the sump water prior to reaching the pump. This increases the NPSH available for the RSS pumps by lowering the vapor pressure of the fluid entering the pumps.

Consistent with Regulatory Guide 1.82 requirements, parametric studies have been performed to bias inputs to minimize containment pressure while maximizing sump water temperature (and vapor pressure).

Unit 2

At Unit 2, the current licensing basis calculations for available NPSH assume that the vapor pressure of the water entering the RSS pump is equal to the containment total pressure. Thus, Unit 2 does not take credit for containment overpressure. This is in accordance with the Standard Review Plan guidance in Section 6.2.2 (Reference 7).

With this assumption, the equation for available NPSH reduces to:

$$NPSH_a = H_{static} - H_{loss}$$

The terms here are consistent with the above definitions.

In FENOC letter L-05-146 (Reference 3), it was stated that it would be necessary to credit containment overpressure for Unit 2 in order to obtain additional NPSH margin. However, the revised analysis supporting this LAR shows that this is not necessary. Therefore each unit's current licensing basis methodology, as described previously, will be maintained. Implementation of this methodology with the MAAP-DBA code was recently reviewed and approved by the NRC in Amendments 271 and 153 (Reference 8).

Proposed Modification

A modification to the containment sump was identified in the FENOC supplemental response to Generic Letter 2004-02 in FENOC letter L-06-020 dated April 3, 2006 (Reference 5). Since the adoption of a licensing basis consistent with Generic Letter 2004-02 will result in an increase in the containment sump strainer head loss, it is necessary to increase the static height of water in the containment sump at both units in order to increase NPSH margin available. This will be accomplished by changing the start signal for the RSS pumps from a fixed time delay from a timer to an ESFAS signal based on a RWST Level Low coincident with a Containment Pressure High-High signal. Starting the RSS pumps on this coincident signal provides better assurance of sump water level at RSS pump start over the range of potential break sizes and single failure assumptions since a fixed amount of water will be transferred from the RWST to the containment. The higher water level will also ensure that the new containment sump strainers will be submerged while accommodating a substantial increase in available surface area.

Methodology Change

FENOC submitted a License Amendment Request in letter L-02-069 dated June 5, 2002 (Reference 9) requesting selective implementation of the Alternative Source Term (AST) methodology. By letter dated September 10, 2003, the NRC approved this request in Amendments 257 and 139 (Reference 10). Subsequently, FENOC implemented AST for LOCA and Locked Rotor Accident based on this approval. In Reference 9 and the subsequent NRC SER (Reference 10), it was stated that inputs (containment pressure, temperature, and condensation rates) to the aerosol removal calculations using the SWNAUA code were calculated using the LOCTIC code.

For this submittal, FENOC used the MAAP-DBA code to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code. Use of the MAAP-DBA code was approved for use at BVPS by Amendments 271 and 153 (Reference 8). Thus, the MAAP-DBA code is the licensing basis code being used for BVPS containment pressure and temperature calculations. The MAAP-DBA code conservatively calculates containment pressure and temperature and condensation rates and is consistent with the approach used in LOCTIC in this regard. The change from LOCTIC to MAAP-DBA to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code is the change in the dose calculation methodology that requires NRC approval.

4.0 TECHNICAL ANALYSIS

The RSS is the primary method of heat removal from the containment environment following a LOCA. As a result, the proposed change to the RSS pump start signal impacts several areas of the BVPS design analyses. These are discussed in the following sections.

Since the proposed change will result in a delay in starting the RSS pumps following a LOCA, heat removal from the sump water and containment will be delayed. This results in a higher containment pressure and temperature profile following the initial peak temperature. This has an impact on the equipment qualification profiles. However, peak containment temperature and pressure are not affected by the proposed change since the peaks occur prior to startup of the RSS pumps.

The change in containment sump water levels and conditions affects the available NPSH analysis for the RSS and LHSI (Unit 1) pumps. The proposed change will result in higher sump water temperatures when transfer to recirculation occurs. The higher sump water temperatures impact piping structural analyses for the lines carrying this water.

The delay in RSS pump start also impacts the LOCA dose consequence analyses since containment atmosphere scrubbing via the RSS will be delayed. Additionally, the installation of the new containment sump strainer increases the metal heat sink mass and surface area in containment. This will impact the containment backpressure used in the 10 CFR 50.46 LOCA analyses.

The proposed change to the RSS pump start signal will also impact the timing of emergency diesel generator loading since the RSS pumps will start later.

4.1 Applicable RSS Pump Start Analysis

Calculations were performed to determine the impact of the proposed change to RSS pump start on the containment pressure and temperature response. A complete set of break sizes and locations was analyzed. This included large and small break LOCA and the spectrum of main steam line breaks (MSLB). These calculations were performed using the MAAP-DBA code consistent with the current BVPS licensing basis. The results of these calculations, summarized in the following sections, demonstrate that all acceptance criteria are met.

In addition to input changes that reflect the strainer heat sink metal mass and surface area, as well as the RSS pump start signal, other inputs were modified to increase operating margin. The QSS and RSS minimum performance curves were reduced to allow for more operating margin for minimum pump performance. The modeling of operator actions to control containment pressure and temperature in the long term was enhanced. This was required since the design of the new containment sump strainers requires limitations on the minimum sump temperature in order to limit the maximum pressure differential. Lower containment sump temperature increases the pressure differential across the strainer due to the higher viscosity of the fluid.

4.1.1 Containment Analysis

The peak containment pressure remains below the design limit of 45 psig. For both units, a LOCA results in the maximum peak containment pressure. For Unit 1, the maximum value was calculated to be 43.1 psig. For Unit 2, the maximum value was calculated to be 44.8 psig. At both units, the maximum peak containment pressure decreased from the current licensing basis. This was expected since the peak containment pressure occurs prior to activation of any sprays and the additional heat sink mass of the new containment strainer provides a small benefit.

The peak containment pressure which results from a MSLB remained the same at Unit 1 and increased slightly (0.1 psig) at Unit 2. However, the results from a MSLB at both units are bounded by the peak containment pressure resulting from a LOCA. The analysis continues to assume no credit for RSS mitigation of a MSLB and therefore the pump start signal modification has no impact on the transient. Plots of the new containment pressure and temperature responses are provided in Attachment D.

4.1.2 NPSH Analysis

Calculations were performed to determine the impact of the proposed change to RSS pump start signal and installation of the containment sump strainer on the NPSH margin for the RSS and LHSI (Unit 1) pumps. Details of NPSH analysis are provided in Attachment D. The methodology for calculating available NPSH is as noted previously and is consistent with the current licensing basis. The calculations establish the available margin in available NPSH without taking into consideration the containment sump strainer head loss. As per Generic Letter 2004-02, this margin will be compared to the head loss across the containment sump strainer following completion of strainer testing with the plant specific debris loading.

Unit 1

At Unit 1, the results show a reduction in the NPSH available for the Inside Recirculation Spray (IRS) pump and an increase for the Outside Recirculation Spray (ORS) pumps. This is due to a more accurate modeling of the direct QSS flow to the pumps. The direct QSS flow to the pumps is configured such that two thirds of the flow goes to the ORS pump and one third to the IRS pump on each train. This arrangement compensates for the higher suction piping loss in the ORS pump. In the current analysis, the enhancement flow was assumed to be equally divided to the IRS and ORS pump. For the IRS pumps, the available NPSH decreased from 17.6 feet to 15.4 feet. For the ORS pumps, the available NPSH increased from 11.8 feet to 14.7 feet. The current NPSH analysis includes a screen head loss component although it is negligible based on the current 50% blockage criteria. The analysis of the modified system operation does not include a strainer head loss component. The required NPSH for all RSS pumps is 9.8 feet and therefore the margin in available NPSH is 5.6 feet for the IRS pumps and 4.9 feet for the ORS pumps based on the modified system operation.

Based on the proposed change to the start signal for the RSS pumps, the minimum water level in the containment sump at RSS pump startup will increase. The RSS pumps will start when the level in the RWST reaches a nominal setpoint of 27 feet

7.5 inches. The limiting cases for minimum containment sump water level are typically smaller breaks where some water inventory remains in the RCS. On RSS pump start, there is a slight reduction in containment water level as the system piping is filled. For the limiting case, the minimum water level above the bottom of the sump increases from 1.25 feet to 4.0 feet. This provides some benefit in available NPSH. The increased water level is needed primarily so that a strainer designed with a surface area larger than the existing screen can be submerged when flow is initiated.

The LHSI pumps at Unit 1 re-align to take flow from the containment sump in the recirculation mode of safety injection. The transfer setpoint is currently approximately 14.0 feet and will not change due to the proposed change to RSS pump start. Transfer to cold leg recirculation will occur after RSS pump start based on the transfer setpoint. Since containment overpressure is credited for the LHSI pump, the available NPSH is sensitive to the containment sump temperature. The proposed RSS start signal change causes the RSS pumps to start later during the transient and consequently less heat removal occurs and a higher sump temperature results at transfer to recirculation. This causes a reduction in the available NPSH for the LHSI pump. The available NPSH for the LHSI pump will be reduced from 27.2 feet to 16.0 feet. The required NPSH is 10.6 feet and therefore the NPSH margin is 5.4 feet. These results establish an upper limit for design of the new containment sump strainer for Unit 1 in terms of available water level at RSS pump start and available margin for head loss (4.9 feet) across the debris laden strainer.

Unit 2

At Unit 2, the increased water level at RSS pump startup increases the available NPSH. Since no containment overpressure is credited, the increased water level has a direct effect on the available NPSH. For the most limiting case, the available NPSH increased from 15.0 feet to 20.0 feet. The current NPSH analysis includes a screen head loss component although it is negligible based on the current 50% blockage criteria. The analysis of the modified system operation does not include a strainer head loss component. The required NPSH is 15.0 feet and therefore the margin in available NPSH is 5.0 feet based on the modified system operation. The minimum containment sump level increases from 2.1 feet to 6.7 feet. These results are based on starting the RSS pumps at a nominal setpoint of 32 feet 9 inches on the RWST. The nominal setpoint for transfer to recirculation is 31.7 feet and will not change due to the proposed change to RSS pump start. This allows for a minimum of approximately 4 minutes of operation of the RSS pumps prior to transferring to recirculation. On transfer to recirculation, two of the four RSS pumps re-align to supply the safety injection header and HHSI pump suctions. Operation in this manner will provide adequate time for the RSS piping to fill and purge any air from the piping prior to transferring. These results establish an upper limit for design of the new containment sump strainer for Unit 2 in terms of available water level at RSS pump start and available margin for head loss (5.0 feet) across the debris laden strainer.

4.1.3 LOCA Analysis

The proposed change to the RSS pump start signal and containment sump strainer impacts the LOCA analyses in several areas. The increase in heat sink inventory associated with the physical strainer installation can cause a decrease in the containment backpressure associated with a LOCA. The change in the RSS pump start will cause a delay in RSS operation which will result in an increase in containment backpressure during the period in which the sprays would have been operating without the proposed delay. Delayed RSS operation also results in an increase in containment sump water temperature at transfer to the cold and hot leg recirculation modes of operation.

An evaluation was performed to determine the impacts of the containment sump modification and the RSS pump start change on the LOCA analyses. The physical structure of the installed strainer represents an increase in the metal heat sink inventory in containment of approximately 1%. This increase in the metal heat sink mass and surface area was used to determine the change in containment backpressure following a LOCA. This analysis did not consider operation of the RSS pumps since with the modified start signal, the pumps will not operate before approximately 20 minutes following initiation of the event. This is well after Peak Cladding Temperature (PCT) occurs and beyond when cladding oxidation is a concern. The results of this evaluation determined that no changes in PCT or cladding oxidation occur as a result of the small change in heat sink inventory and the RSS pump start signal modification.

The sump water temperature was calculated at transfer to safety injection cold leg recirculation and transfer to hot leg recirculation. The maximum increase in sump water temperature at transfer to cold leg recirculation was calculated to be approximately 40°F. The maximum sump temperature remains bounded by the current assumptions in the LOCA calculations. The sump water temperature at transfer to hot leg recirculation will increase by approximately 2°F. This increase is also bounded by the current analysis assumptions.

4.1.4 Dose Analyses

The dose consequences analyses for a LOCA are impacted by the proposed RSS modifications. The major effect involves the amount and timing of aerosol removal from the containment atmosphere which affects the doses associated with containment leakage. This is due to the additional delay in starting the RSS pumps. The Engineered Safety Feature leakage and RWST back leakage contributions are also impacted by the delay in initiation of recirculation spray including the change in containment sump water volume and temperature. A re-analysis of the low population zone (LPZ), exclusion area boundary (EAB), emergency response facility (ERF), and control room doses was performed. This re-analysis was done consistent with the methodology used in the current licensing basis calculation with the following exception. The MAAP-DBA code was used to calculate containment pressure, temperature and condensation rate inputs to the SWNAUA code which is used to calculate the aerosol removal rates. The results show small increases in all reported doses; the EAB dose increases from 14 to 16.5 TEDE (rem), the LPZ dose increases from 2.5 to 3.0 TEDE (rem), the ERF dose increases from 3.0 to 3.5 TEDE (rem) and the control room dose increases from 2.0 to 2.5 TEDE (rem). A more detailed description of the analyses is provided in Attachment E.

4.1.5 Piping Analysis

As stated previously, the sump water temperature will be increased for both the cold leg and hot leg recirculation modes. Therefore, the piping in these flow paths will be analyzed at the new higher values.

At Unit 1, analyses will be completed and any required pipe support modifications will be installed prior to operation with the proposed RSS pump start change.

For Unit 2, no analyses are required since the current analyses bound operating parameters which result from the proposed RSS pump start change. At Unit 2, two of the four RSS pumps re-align to provide flow to the safety injection header and the HHSI pumps. This path also includes a RSS heat exchanger which cools the water prior to entering the Safety Injection System. The current Unit 2 analysis is based on a failure of service water to one train of RSS heat exchangers. This operating condition will be alleviated by revising procedures to instruct the operators to secure a train of RSS if no service water is available for that train. Operating a train of RSS without cooling water only adds to the containment heat load. There is adequate time to secure the RSS train without service water since, with the proposed change, the earliest the RSS pumps would start at Unit 2 is approximately 40 minutes after the event, whereas service water flow to the RSS heat exchanger initiates following a CIB signal which occurs much earlier.

4.1.6 Equipment Qualification Analysis

In the short term (up to 1000 seconds), the Equipment Qualification (EQ) temperature profile in containment is governed by the MSLB. The analysis results indicate that a small increase in the peak containment temperature occurs following a MSLB. The maximum containment temperature is slightly above (less than 1°F) the current EQ envelope. This is due to an increase in the maximum initial containment temperature which was assumed in the new analysis. The purpose of the increase maximum initial containment temperature is to allow for an increase in the containment temperature operating envelope. The value of the maximum initial containment temperature assumed in the revised containment analysis (108°F), results in a proposed change to the current Technical Specification upper limit for containment temperature.

The longer term EQ temperature profile (> 1000 seconds) is governed by a LOCA event. The containment temperature following a LOCA increases due to the proposed delay in starting the RSS pumps. This is expected since the RSS provides the mechanism for removing heat from the containment. Again, the accident temperature profile slightly exceeds (less than 2°F) the existing EQ profile for a brief period at approximately 3 hours following accident initiation. Attachment D provides more details of this evaluation.

Analysis of the impact of the increased EQ profile is ongoing and will be completed prior to approval of this LAR and the subsequent operation with the proposed change to the RSS pump start signal. It is not expected that the change in containment temperature will have a significant impact on equipment qualification. However, any equipment that must be replaced or re-qualified will be addressed prior to operation with the proposed change to RSS pump start.

4.1.7 Emergency Diesel Generator (EDG) Loading

With the proposed RSS pump start change, the inside containment RSS pumps at Unit 1 will start immediately following receipt of an RWST Level Low signal coincident with a Containment Pressure High-High signal. The Unit 1 outside containment RSS pumps will be started following a 15 second delay after receiving this coincident signal. The 15 second delay limits the starting load on the EDG and maintains a staggered pump start timing. The maximum loading on the Unit 1 EDG will not increase as a result of this modification. The maximum load will now occur at a later time due to the delay in starting the RSS pumps and will not violate diesel loading criteria.

With the proposed RSS pump start change, all four RSS pumps at Unit 2 will start immediately following receipt of an RWST Level Low signal coincident with a Containment Pressure High-High signal. The maximum load on the EDG will not increase as a result of this modification but will occur at a later time due to the delay in starting the RSS pumps.

4.1.8 Start Signal Reliability and Accuracy

The reliability of the proposed change to the RSS pumps start signal compared to the existing time delay start is only slightly decreased. For the case when all four vital bus power supplies are available to provide the 2-of-3 RWST Level Low signals, there is an increase of approximately 1.2E-03 in the probability of a RSS pump not automatically starting following a coincident RWST Level Low/Containment Pressure High-High signal. These proposed modifications result in approximately a 15% increase at Unit 1, and approximately a 10% increase at Unit 2, in the failure probability of a single RSS pump train.

Taking credit for operators to manually start the pump would reduce this increase in failure probability by about 2 orders of magnitude. In addition, the RWST level transmitters fail low upon loss of vital bus power to the instrument loop, so this increase in the signal failure probability would be even slightly lower than the above values, given the failure of a vital bus to power the instrument loop. When considering common cause failures of both trains of signals, the increase in the conditional probability of the Train B RSS pumps failing due to the RWST Level Low signal, given that Train A has probabilistically failed, would be approximately 1.3E-01 when all four vital bus power supplies are available. This would reduce to approximately 1.0E-01, if there were a failure of a vital bus to power one of the instrument loops. If the Train A signal was successful, the increase in the Train B RSS pump conditional failure probability due to the Train B RWST Level Low signal would be approximately 1.1E-03 with or without vital bus failures.

The RWST Level Low instrumentation is the same as the RWST Level Extreme Low instrumentation with the exception of the RWST Level Low comparator which is physically distinct but of the same type as the RWST Level Extreme Low comparator. Therefore the accuracy of the RWST Level Low actuation signal is identical to the RWST Level Extreme Low accuracy. The RWST Level Low instrument uncertainty was developed in accordance with Westinghouse Setpoint Methodology for Protection Systems WCAP-11419 Rev. 5 for Unit 1 (Reference 11) and WCAP-11366 Rev. 7 for Unit 2 (Reference 12).

4.1.9 Core Damage Frequency and Large Early Release Frequency Effects

The increase in the core damage frequency above the baseline values due to the implementation of this proposed modification would be less than 5.0E-08 per year for each unit, which is considered to be of very low risk significance. The increase in large early release frequency would also be insignificant (less than 1.0E-11 per year), due to the dominating steam generator tube rupture and interfacing systems LOCA contributions, which bypass containment.

4.1.10 Other Safety Analyses

No other safety analyses are impacted by the proposed change to RSS pump start or the containment sump modification. The analysis of MSLB in containment does not credit operation of the RSS pumps and therefore delaying the start has no impact. None of the non-LOCA transients credit or are affected by the RSS pumps.

4.2 Visual Inspection Surveillance

The proposed SR 3.5.2.7 changes reflect the modification to the ECCS containment sump suction inlet strainers. Specifically, the ECCS containment sump suction inlets are being changed from a trash rack and screen-type arrangement to a larger strainer-type inlet. The requirements for inspection of the sump portion of the ECCS system will remain essentially the same as currently required by SR 3.5.2.7. Specifically, the inspection requirements will continue to state that the ECCS containment sump section inlet is to be visually inspected at an 18 month frequency to verify that it is not restricted by debris and shows no evidence of structural distress or abnormal corrosion.

The proposed changes modify the requirements of SR 3.5.2.7 by removal of the word "train", changing the terminology of "trash racks and screens" to "strainers" consistent with the new design, and adding the wording "accessible regions" to describe the extent of the visual inspection. The "accessible regions" of the sump strainers are those regions that can be visually examined without disassembling the strainer assembly or the grating and cover plates over the strainer assembly.

The removal of the word "train" is considered to be a clarification of the current SR because the sump strainer is a combined header for both ECCS trains; hence, use of the word "train" is not needed.

The terminology change from "trash racks and screens" to "strainers" provides a more appropriate, and generic, description of the new configuration that strains through perforated stainless steel plates. The modifications to the containment sump suction inlets are being done in accordance with evaluations that have been or are being done in response to NRC Generic Letter 2004-02 (Reference 3) and the FENOC engineering change process. The new strainers are functionally equivalent to the trash racks and screens for meeting the requirements of 10 CFR 50.46(b)(5) for long term cooling. The use of the term "strainers" will not affect the implementation of SR 3.5.2.7.

The addition of "accessible regions" to the SR is due to the size, complexity and location of the new containment sump strainers. The accessible regions of the strainers are those areas that can be accessed by an inspector without disassembling the strainer unit or the protective grating and plates over the strainers. The new sump strainers consist of a much larger strainer area than that of the existing sump screens. Grating and solid plates are located above the strainers to comply with recommendation eight of Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

At Unit 2, where one hundred thirteen (113) vertical top-hat strainer units have already been installed, the grating protects the strainers and provides a platform for access over the strainers. A similar multi-strainer installation is planned for Unit 1. Each row of strainers is two to five strainers deep. Since the grating is located only a few inches over the top of the strainers, only the front side of the exterior row of strainers is accessible for visual inspection. The position of the platform and the top of the strainers does not provide enough clearance to allow an inspector access to the internal strainers. The platform is built in sections that are bolted in place and would require rigging equipment to handle if lifted. Because of the larger strainer area, the strainer surface area approaches the containment wall on one side of the array. As a result, there is insufficient space between the strainer and the containment wall on the "back" side of the strainer assembly to allow an inspector access. Therefore, access to the strainer surface area is available from only the three "exterior" sides of the array.

The grating and solid cover plates, while presenting a challenge to inspection; also serve to keep the strainers free from debris during normal plant operation by preventing foreign material from falling into the strainers. The strainers are carefully installed with cleanliness checks throughout installation. Once the grating and plates over the strainers are in place, there is no mechanism during normal operation to foul the strainers with debris. Inspecting the entire strainer surface area each outage would put the strainers at risk of damage when the grating over the strainers was disassembled and hoisted out of the way to allow inspection of the covered area. Limiting the visual inspection to only the "accessible regions" will also help minimize the dose to the individuals performing the inspection. Since the strainers are constructed of stainless steel, it is very unlikely that they would corrode. An exterior inspection of the accessible areas will assure that the strainer units remain clean and will allow the monitoring of the strainers for structural distress and abnormal corrosion.

5.0 REGULATORY SAFETY ANALYSIS

In this License Amendment Request (LAR) the Beaver Valley Power Station (BVPS) Units 1 and 2 Technical Specifications are being revised to be consistent with a proposed change to the Recirculation Spray System (RSS) pump start signal due to a modification to the containment sump screens. The first Technical Specification change consists of starting the RSS pumps on a coincident Containment Pressure High-High/Refueling Water Storage Tank (RWST) Level Low Engineered Safety Feature Actuation System (ESFAS) signal instead of a signal from a timer that imposes a fixed time delay following a Containment Pressure High-High signal. The existing timer imposes a fixed time delay on the pump start following a Containment Pressure High-High signal. The proposed change to the pump start signal results in the pump start being driven by plant conditions. The resultant extended pump start assures that plant conditions are such that the pumps are started when needed and that sufficient water is present in the containment sump. The second Technical Specification change consists of modifying the nomenclature for the ESFAS signal used for automatic switchover of Emergency Core Cooling System (ECCS) suction to the containment sump. The third Technical Specification change consists of revising the containment sump 18 month visual inspection Surveillance Requirement. The fourth Technical Specification change consists of raising the upper limit on containment average air temperature to reflect a revised containment analysis. The fifth Technical Specification change consists of lowering peak containment pressure, P_a, to reflect a revised containment analysis.

Additionally, Nuclear Regulatory Commission (NRC) approval of a change to the calculation methodology used to determine aerosol removal coefficients for use in dose consequence analyses is requested. The methodology change consists of using MAAP-DBA instead of LOCTIC to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code. All of these changes are being made to support BVPS resolution of NRC Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance."

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 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No. The proposed changes to the Recirculation Spray System (RSS) pump start signal, the upper containment temperature Technical Specification (TS) limit, the peak containment internal pressure, the nomenclature for automatic switchover to the containment sump and the containment sump screen visual inspection surveillance requirement do not involve any system or component that are accident initiators. The RSS is used for accident mitigation only. The Refueling Water Storage Tank (RWST) level and containment pressure instrumentation will continue to comply with all applicable regulatory requirements and design criteria (e.g., train separation, redundancy, single failure, etc.) following approval of the proposed changes. The design functions performed by the RSS and the containment are not changed by this license amendment request.

Delaying the start of the RSS pumps and the change to the upper containment temperature affect the long-term containment pressure and temperature profiles. The environmental qualification of safety-related equipment inside containment will be confirmed to be acceptable and accident mitigation systems will continue to operate within design temperatures and pressures. Delaying the RSS pump start reduces the emergency diesel generator loading in the early stage of a design basis accident and maintaining the staggered loading of the RSS pump starts avoids overloading on each emergency diesel generator at Unit 1. Staggered loading of the emergency diesel generator is not required for Unit 2.

The methodology change to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code will not involve a significant increase in the probability of an accident previously evaluated because this change in methodology does not impact accident initiators.

The loss of coolant accident (LOCA) has been evaluated using the guidance provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The radiological consequences of the remaining design basis accidents are not significantly impacted by the proposed changes. As demonstrated by the supporting analyses, the estimated dose consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room remain within the acceptance criteria of 10 CFR 50.67 as supplemented by Regulatory Guide 1.183 and Standard Review Plan Section 15.0.1. In addition, the supporting analyses also demonstrates that the dose consequences in the Emergency Response Facility remain compliant 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. The revised radiological analyses results in a slight increase in control room and off-site doses; however, the radiological analyses and evaluations developed in support of this application demonstrate that the proposed changes will not impact compliance with applicable regulatory requirements and will not involve a significant increase in the consequences of an accident previously evaluated. The slight increase in control room and off-site doses is more than offset by the increased assurance of adequate NPSH to the RSS pumps and Emergency Core Cooling System operability.

The safety analysis acceptance criteria will continue to be met following the proposed changes to the RSS pump start signal, visual sump inspection, TS containment upper temperature limit, peak containment internal pressure, nomenclature for automatic switchover to the containment sump and the change to the control room and off-site dose consequences analyses.

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

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

   Response: No. One of the proposed changes alters the RSS pump start circuitry by initiating the pump start from a coincident Containment Pressure High-High/RWST Level Low signal instead of from a timer. The RSS pump instrumentation will be included as part of the Engineered Safety Feature Actuation System (ESFAS) instrumentation in the TS and will be subject to the ESFAS surveillance requirements following approval of the proposed changes. The design of the RSS pump start instrumentation complies with all applicable regulatory requirements and design criteria. The failure modes have been analyzed to ensure that the revised RSS pump start circuitry can withstand a single active failure without affecting the RSS design functions. The RSS is an accident mitigation system only, so no new accident initiators are created.

It is not expected that the change in containment temperature will have a significant impact on equipment qualification. However, any equipment that must be replaced or re-qualified will be addressed prior to operation with the proposed change to RSS pump start. As a result any such equipment will not introduce new failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident.

The remaining changes do not change plant equipment design or function and therefore will not introduce new failure modes, accident initiators, or malfunctions that would cause a new or different kind of accident.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No. The changes to the RSS pump start signal and the upper containment temperature limit affect the containment response and the LOCA dose analyses. Analyses demonstrate that containment design basis limits are satisfied and post-LOCA offsite and control room dose criteria will continue to be met following approval of the proposed changes.

The change to the containment sump visual inspection will not involve a significant reduction in a margin of safety because the revised surveillance will continue to provide adequate assurance the sump screens are not blocked with debris and that signs of corrosion will be detected.

The change to peak containment internal pressure will not result a significant reduction in a margin of safety because the new pressure is lower for each of the units.

Although the control room and off-site doses slightly increase (due to a combination of the change to the start signal and the proposed methodology change), the increase will not involve a significant reduction in a margin of safety because operator and public exposure limits will continue to meet applicable regulatory requirements.

Therefore, the proposed changes do 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 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

A review of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (Reference 13) and the Regulatory Guides, was conducted to assess the potential impact associated with the proposed changes. The General Design Criteria (GDC) and the Regulatory Guides (RG) were evaluated as follows:

General Design Criterion 16, with respect to containment design.

General Design Criterion 35, with respect to emergency core cooling.

General Design Criterion 36, with respect to inspection of emergency core cooling system.

General Design Criterion 37, with respect to testing of emergency core cooling system.

General Design Criterion 38, with respect to the RSS.

General Design Criterion 39, with respect to permitting periodic inspection of the RSS.

General Design Criterion 40, with respect to permitting periodic testing of the RSS.

General Design Criterion 41, as it relates to the control of fission products.

General Design Criterion 42, with respect to inspection of containment atmosphere cleanup systems.

General Design Criterion 43, with respect to testing of containment atmosphere cleanup systems.

General Design Criterion 50, with respect to the containment design basis.

General Design Criterion 51, with respect to the fracture prevention of the containment pressure boundary.

Regulatory Guide 1.1, as it relates to the NPSH available to the RSS pumps.

Regulatory Guide 1.26, "Quality Group Classifications and Standards", as it relates to quality group standards for the RSS.

Regulatory Guide 1.29, "Seismic Design Classification", as it relates to seismic classification of the RSS.

Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems", as it relates to the design of the containment sump.

Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", as it relates to dose consequences.

Assessment

No change to either unit's Updated Final Safety Analysis Report (UFSAR) description of conformance to the GDCs or the listed Regulatory Guides is required as a result of the changes proposed in this LAR.

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 amendments would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendments do 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 amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

7.0 PRECEDENT

Other license amendment requests have been submitted to resolve the concerns of GSI-191 (Reference 1) and Generic Letter 2004-02 (Reference 4). Some of the most recent submittals were made by Virginia Electric and Power Company, Dominion Nuclear Connecticut, Inc. and TXU Generation Company LP (TXU Power). The Virginia Electric and Power Company submittals were made for North Anna Power Station Units 1 and 2 (Reference 14) and Surry Power Station Units 1 and 2 (Reference 15). The Dominion Nuclear Connecticut, Inc. submittal was made for Millstone Power Station Unit 3 (Reference 16). The TXU Power submittal was made for the Comanche Peak Steam Electric Station Units 1 and 2 (Reference 17). The Surry license amendment requests were approved by the NRC with the issuance of license amendments 250 and 249 (Reference 18). The Comanche Peak license amendment request was approved by the NRC with the issuance of license amendments 129 for Units 1 and 2 (Reference 19). The Millstone Unit 3 license amendment request was approved by the NRC with the issuance of license amendment 233 for Unit 3 (Reference 20).

Each of these submittals except Comanche Peak requested NRC approval to change the start signal for the recirculation spray pumps from a timer to an ESFAS signal. The Surry and North Anna submittals proposed a coincident containment pressure/RWST level ESFAS signal, whereas the Millstone Unit 3 submittal proposed only an RWST level ESFAS signal. Each of these license amendment requests also submitted revised safety analyses to support the proposed change to the recirculation spray pumps start signal. In this regard, the BVPS submittal is comparable to these submittals. The Comanche Peak submittal proposed a reduced Allowable Value setpoint for RWST level.

8.0 REFERENCES

1. NRC Generic Safety Issue 191 (GSI-191), “Assessment of Debris Accumulation on PWR Sump Performance.”

2. FENOC letter L-05-027, “License Amendment Request Nos. 296 and 169,” dated February 25, 2005.

3. FENOC letter L-05-146, “Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,"” dated September 6, 2005.

4. Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” September 13, 2004.

5. FENOC letter L-06-020, “Supplemental Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,"” dated April 3, 2006.

6. FENOC letter L-06-145, “Revised Commitment Dates Relevant to FirstEnergy Nuclear Operating Company Correspondence to the NRC, L-06-020, dated April 3, 2006,” dated September 29, 2006.

7. Standard Review Plan, Section 6.2.2, Containment Heat Removal Systems.

8. NRC letter dated February 6, 2006, “BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2) – ISSUANCE OF AMENDMENTS RE: CONTAINMENT CONVERSION FROM SUBATMOSPHERIC TO ATMOSPHERIC OPERATING CONDITIONS (TAC NOS. MC3394 AND MC3395,” Amendments 271 and 153.

9. FENOC letter L-02-069, "License Amendment Request Nos. 300 and 172," dated June 5, 2002.

10. NRC Letter date September 10, 2003, "BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT RE: SELECTIVE IMPLEMENTATION OF ALTERNATE SOURCE TERM AND CONTROL ROOM HABITABILITY TECHNICAL SPECIFICATION CHANGES (TAC NOS. MB5303 AND MB5304," Amendments 257 and 139.

11. WCAP-11419, Rev. 5, "Westinghouse Setpoint Methodology for Protection Systems – Beaver Valley Power Station Unit 1," May 2005.

12. WCAP-1136, Rev. 7, "Westinghouse Setpoint Methodology for Protection Systems – Beaver Valley Power Station Unit 2," April 2005.

13. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

14. Virginia Electric and Power Company, Serial 06-849, dated October 3, 2006, "VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION) NORTH ANNA POWER STATION UNITS 1 AND 2 PROPOSED TECHNICAL SPECIFICATION CHANGE AND SUPPORTING SAFETY ANALYSES REVISIONS TO ADDRESS GENERIC SAFETY ISSUE 191."

15. Virginia Electric and Power Company, Serial 06-014, dated January 31, 2006, "VIRGINIA ELECTRIC AND POWER COMPANY SURRY POWER STATION UNITS 1 AND 2 PROPOSED TECHNICAL SPECIFICATION CHANGE AND SUPPORTING SAFETY ANALYSES REVISIONS TO ADDRESS GENERIC SAFETY ISSUE 191."

16. Dominion Nuclear Connecticut, Inc., Serial 05-357, dated September 13, 2005, "DOMINION NUCLEAR CONNECTICUT, INC. MILLSTONE POWER STATION UNIT 3 PROPOSED TECHNICAL SPECIFICATIONS CHANGE RECIRCULATION SPRAY SYSTEM."

17. TXU Power, Log# TXX-05199, dated December 16, 2005, "LICENSE AMENDMENT REQUEST (LAR) 05-010, REVISION TO TECHNICAL SPECIFICATION 3.3.2, "ESFAS INSTRUMENTATION," 3.5.2, "ECCS-OPERATING," AND 3.6.7, "SPRAY ADDITIVE SYSTEM."

18. NRC Letter dated October 12, 2006, "SURRY POWER STATION UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING IMPLEMENTATION OF GENERIC SAFETY ISSUE 191 (TAC NOS. MC9724 AND MC9725," Amendment Nos. 250 and 249.

19. NRC Letter dated October 5, 2006, "COMANCHE PEAK STEAM ELECRIC STATION (CPSES) UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATION 3.3.2, "ESFAS [ENGINEERED SAFETY FEATURES ACTUATION SYSTEM] INSTRUMENTATION," AND 3.5.2, "ECCS [EMERGENCY CORE COOLING SYSTEM] -OPERATING," (TAC NOS. MC9494 AND MC9495," Amendment Nos. 129 and 129.

20. NRC Letter dated September 20, 2006, "MILLSTONE POWER STATION, UNIT NO. 3 – ISSUANCE OF AMENDMENT RE: RECIRCULATION SPRAY SYSTEM," (TAC NO. MC98327," Amendment No. 233.

Enclosure Attachment A

Beaver Valley Power Station, Unit Nos. 1 and 2
Proposed Technical Specification Changes

License Amendment Requests 334 (Unit 1) and 205 (Unit 2)

The following is a list of the affected pages:

Page		Page
3.3.2 - 1 *		3.5.2 - 1 *
3.3.2 - 2 *		3.5.2 - 2 *
3.3.2 - 3 *		3.5.2 - 3
3.3.2 - 4 *		
3.3.2 - 5 *		3.6.5 - 1
3.3.2 - 6 *		
3.3.2 - 7 *		5.5 - 18 *
3.3.2 - 8		5.5 - 19
3.3.2 - 9		
3.3.2 - 10 *		
3.3.2 - 11 *		
3.3.2 - 12		
3.3.2 - 13 *		

* No Change. Page provided for context only.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Be in MODE 5.	84 hours
C. One train inoperable.	**- NOTE -** One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE.	
	C.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2.1 Be in MODE 3.	12 hours
	AND	
	C.2.2 Be in MODE 5.	42 hours
D. One channel inoperable.	- NOTE - The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.	
	D.1 Place channel in trip.	6 hours
	OR	
	D.2.1 Be in MODE 3.	12 hours
	AND	
	D.2.2 Be in MODE 4.	18 hours
E. One Containment Pressure channel inoperable.	- NOTE - One additional channel may be bypassed for up to 4 hours for surveillance testing.	
	E.1 Place channel in bypass.	6 hours
	OR	
	E.2.1 Be in MODE 3.	12 hours
	AND	
	E.2.2 Be in MODE 4.	18 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	F.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	F.2.2 Be in MODE 4.	60 hours
G. One train inoperable.	**- NOTE -** One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE.	
	G.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	G.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	G.2.2 Be in MODE 4.	18 hours
H. One channel inoperable.	**- NOTE -** The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.	
	H.1 Place channel in trip.	6 hours
	<u>OR</u>	
	H.2 Be in MODE 3.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One Main Feedwater Pumps trip channel inoperable.	I.1 Restore channel to OPERABLE status.	48 hours
	OR	
	I.2 Be in MODE 3.	54 hours
J. One channel inoperable.	- NOTE - One additional channel may be bypassed for up to 4 hours for surveillance testing.	
	J.1 Place channel in bypass.	6 hours
	OR	
	J.2.1 Be in MODE 3.	12 hours
	AND	
	J.2.2 Be in MODE 5.	42 hours
K. One or more channels inoperable.	K.1 Verify interlock is in required state for existing unit condition.	1 hour
	OR	
	K.2.1 Be in MODE 3.	7 hours
	AND	
	K.2.2 Be in MODE 4.	13 hours

SURVEILLANCE REQUIREMENTS

- NOTE -
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.3	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.4	Perform COT.	92 days
SR 3.3.2.5	- NOTE - Verification of relay setpoints not required. Perform TADOT.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.6	Perform SLAVE RELAY TEST.	----------------------- **- NOTE -** Only applicable to Unit 2 ----------------------- 92 days OR ----------------------- **- NOTE -** Only applicable to Unit 2 provided a satisfactory contact loading analysis has been completed, and a satisfactory slave relay service life has been established, for the slave relay being tested. ----------------------- 12 Months AND ----------------------- **- NOTE -** Only applicable to Unit 1 ----------------------- 18 Months
SR 3.3.2.7	----------------------- **- NOTE -** Verification of setpoint not required. ----------------------- Perform TADOT.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.8	-- **- NOTE -** This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.2.9	-- **- NOTE -** Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is ≥ 600 psig. -- Verify ESFAS RESPONSE TIMES are within limit.	18 months on a STAGGERED TEST BASIS

Table 3.3.2-1 (page 1 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4[e][f] SR 3.3.2.8[e][f] SR 3.3.2.9	$\le$ 5.33 psig	$\le$ 5.3 psig
d. Pressurizer Pressure - Low	1,2,3[a]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	$\ge$ 1841 psig	$\ge$ 1852 psig
e. Steam Line Pressure - Low	1,2,3[a]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	$\ge$ 495.8 psig with time constant τ_1 $\ge$ 50 secs and $\tau_2 \le$ 5 secs	$\ge$ 494 psig with time constant τ_1 $\ge$ 50 secs and τ_2 $\le$ 5 secs
2. Containment Spray Systems						
a. Quench Spray						
~~a~~(1)~~.~~ Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.7	NA	NA
~~b~~(2)~~.~~ Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA

(a) Above the P-11 (Pressurizer Pressure) interlock.

(e) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(f) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

Table 3.3.2-1 (page 2 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
2. Containment Spray Systems						
e(3). Containment Pressure High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≤ 11.43 psig	≤ 11.4 psig
b. Recirculation Spray						
(1) Automatic Actuation Logics	1,2,3	2 trains	F	SR 3.3.2.2 SR 3.3.2.3	NA	NA
(2) Refueling Water Storage Tank (RWST) Level Low	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f)	≥ 27' 4" and ≤ 27' 11"	≥ 32' 8" and ≤ 32' 10"
Coincident with						
Containment Pressure High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≤ 11.43 psig	≤ 11.4 psig
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.7	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA

(3) Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≤ 11.43 psig	≤ 11.4 psig

(e) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(f) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

No change. Page included for context only.

Table 3.3.2-1 (page 3 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
4. Steam Line Isolation						
a. Manual Initiation (Only applicable to Unit 2)	1,2(b), 3(b)	2 per train, 2 trains	F	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(b), 3(b)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA
c. Containment Pressure - Intermediate High High	1,2(b), 3(b)	3	D	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≤ 7.33 psig	≤ 7.3 psig
d. Steam Line Pressure						
(1) Low	1,2(b), 3(a)(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≥ 495.8 psig with time constant τ_1 ≥ 50 secs and τ_2 ≤ 5 secs	≥ 494 psig with time constant τ_1 ≥ 50 secs and τ_2 ≤ 5 secs
(2) Negative Rate - High	3(b)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8 SR 3.3.2.9	≤ 104.2 psi with a time constant ≥ 50 secs	≤ 103.6 psi with a time constant ≥ 50 secs

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Except when all MSIVs are closed and de-activated.

(c) Below the P-11 (Pressurizer Pressure) interlock when SI on steam line pressure low is blocked.

(e) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(f) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

No change. Page included for context only.

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(d), 3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA
b. SG Water Level - High High (P-14)	1,2(d), 3(d)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≤ 90.2%	≤ 92.7%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.6	NA	NA
b. SG Water Level - Low Low	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f) SR 3.3.2.9	≥ 19.1%	≥ 20%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(d) Except when all Main Feedwater Lines are isolated by either closed and deactivated MFIVs, or MFRVs and associated bypass valves, or closed manual valves.

(e) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(f) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

Table 3.3.2-1 (page 5 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
6. Auxiliary Feedwater						
d. Undervoltage Reactor Coolant Pump	1,2	1 per bus	H	SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 2962 V	≥ 2962 V
e. Trip of all Main Feedwater Pumps	1,2	1 per pump	I	SR 3.3.2.7 SR 3.3.2.9	NA	NA
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic	1,2,3,4	2 trains	B	SR 3.3.2.2 SR 3.3.2.3	NA	NA
b. Refueling Water Storage Tank (RWST) Level Extreme Low ~~(Unit 1)~~ ~~Extreme Low~~ ~~(Unit 2)~~	1,2,3,4	42 trains	J	SR 3.3.2.1 SR 3.3.2.4(e)(f) SR 3.3.2.8(e)(f)	≥ 13' 9" and ≤ 14' 4"	≥ 31' 8" and ≤ 31' 10"
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(e) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

(f) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodology used to determine the Nominal Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.

Table 3.3.2-1 (page 6 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	UNIT 1 ALLOWABLE VALUE	UNIT 2 ALLOWABLE VALUE
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.7	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8	≤ 2004 psig	≤ 2004 psig
c. T_{avg} - Low Low, P-12	1,2,3	1 per loop	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.8	≥ 540.5°F	≥ 540.5°F

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

- NOTES -

1. In MODE 3, both low head safety injection pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. In MODE 3, one of the required charging pumps may be made incapable of injecting to support transition into or from the Applicability of LCO 3.4.12, "Overpressure Protection System (OPPS)," for up to 4 hours or until the temperature of all RCS cold legs exceeds the OPPS enable temperature specified in the PTLR plus 25°F, whichever comes first.

3. For Unit 1 only. In MODE 3, the ECCS automatic high head safety injection (HHSI) flow path may be isolated to support transition into or from the Applicability of LCO 3.4.12, "Overpressure Protection System (OPPS)" for up to 4 hours or until the temperature of all RCS cold legs exceeds the OPPS enable temperature specified in the PTLR plus 25°F, whichever comes first.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator control circuit removed. For Unit 1 only Number / Position / Function MOV-1SI-890A / Closed / Low head safety injection (LHSI) to Hot Leg MOV-1SI-890B / Closed / LHSI to Hot Leg MOV-1SI-890C / Open / LHSI to Cold Leg MOV-1SI-869A / Closed / HHSI Pump to Hot Leg MOV-1SI-869B / Closed / HHSI Pump to Hot Leg For Unit 2 only Number / Position / Function 2SIS*MOV8889 / Closed / LHSI to Hot Legs 2SIS*MOV869A / Closed / HHSI to Hot Leg 2SIS*MOV869B / Closed / HHSI to Hot Leg 2SIS*MOV841 / Open / HHSI to Cold Leg 2CHS*MOV8132A / Open / HHSI Pump Discharge Cross Connect 2CHS*MOV8132B / Open / HHSI Pump Discharge Cross Connect 2CHS*MOV8133A / Open / HHSI Pump Discharge Cross Connect 2CHS*MOV8133B / Open / HHSI Pump Discharge Cross Connect	12 hours
SR 3.5.2.2	Verify the HHSI pump minimum flow valve is open with power to the valve operator removed.	12 hours
SR 3.5.2.3	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.7	Verify, by visual inspection, ~~each~~ that accessible regions of the ECCS ~~train~~ containment sump suction inlet ~~is~~ are not restricted by debris and that the accessible regions of the ~~suction inlet trash racks and screens~~ strainers show no evidence of structural distress or abnormal corrosion.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≥ 70°F and ≤ ~~105°F~~108°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limits.	24 hours

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

a. The SFDP shall contain the following:

1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities, and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

3. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Containment Leakage Rate Testing Program

a. A program shall establish 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. For Unit 1, exemptions to Appendix J of 10 CFR 50 are dated November 19, 1984, December 5, 1984, and July 26, 1995. For Unit 2, exemptions to Appendix J of 10 CFR 50 are as stated in

5.5 Programs and Manuals

5.5.12 Containment Leakage Rate Testing Program (continued)

the Operating License. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

1. For Unit 1, the next Type A test performed after the May 29, 1993 Type A test shall be performed no later than May 28, 2008.

2. For Unit 2, the next Type A test performed after the November 10, 1993 Type A test shall be performed no later than November 9, 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is ~~43.3~~43.1 psig (for Unit 1) and ~~44.9~~44.8 psig (for Unit 2).

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

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup prior to MODE 4 entry following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each emergency air lock door, no detectable seal leakage when gap between door seals is pressurized to $\geq$ 10 psig or door seal leakage quantified to ensure emergency air lock door seal leakage rate is $\leq 0.0005\ L_a$ when tested at $\geq$ 10 psig.

c) For each personnel air lock door, no detectable seal leakage when gap between door seals is pressurized to $\geq P_a$ or door seal leakage quantified to ensure personnel air lock door seal leakage rate is $\leq 0.0005\ L_a$ when tested at $\geq P_a$.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

Enclosure Attachment B

Beaver Valley Power Station, Unit Nos. 1 and 2
Proposed Technical Specification Bases Changes
License Amendment Requests 334 (Unit 1) and 205 (Unit 2)

Technical Specification Bases changes are provided for Information Only

The following is a list of the affected pages:

Page		Page
B 3.3.2 - 7 *		B 3.5.2 - 10
B 3.3.2 - 8 *		B 3.5.2 - 11
B 3.3.2 - 9		
B 3.3.2 - 10 *		B 3.6.1 - 2
B 3.3.2 - 11 *		
B 3.3.2 - 12 *		B 3.6.2 - 2
B 3.3.2 - 13 *		
B 3.3.2 - 14		B 3.6.4 - 1
B 3.3.2 - 15		
B 3.3.2 - 16		B 3.6.5 - 1 *
B 3.3.2 - 17 *		B 3.6.5 - 2
B 3.3.2 - 18 *		
B 3.3.2 - 19 *		B 3.6.6 – 1 *
B 3.3.2 - 20 *		B 3.6.6 - 2
B 3.3.2 - 28		B 3.6.7 - 1
B 3.3.2 - 29 *		B 3.6.7 - 2
		B 3.6.7 - 3
B 3.3.2 - 34		B 3.6.7 - 4
B 3.3.2 - 35 *		
		B 3.6.7 - 8
B 3.3.2 - 38		
B 3.3.2 - 39 *		

No Change. Page provided for context only.

No change. Page included for context only.

Provided for Information Only.

BACKGROUND (continued)

indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the selected decision logic matrix functions and partially test the actuation relays while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays that provide actuation signals to ESF components are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions not explicitly credited in the safety analysis, may be implicitly credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions not explicitly analyzed and may be anticipatory in nature or serve as backups to Functions that are explicitly credited in the accident analysis to provide defense in depth (Ref. 1).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. A channel is OPERABLE provided the trip setpoint "as-found" value does not exceed its associated Allowable Value and provided the trip setpoint "as-left" value is adjusted to a value within the calibration tolerance band of the nominal trip setpoint. A trip setpoint may be set more conservative than the nominal trip setpoint as necessary in response to plant conditions provided that the ± calibration tolerance band remains the same and the Allowable Value is administratively controlled accordingly in the conservative direction to meet the assumptions of the setpoint methodology. The conservative direction is established by the direction of the inequality applied to the Allowable Value. Failure of any instrument may render the affected channel(s) inoperable and reduces the reliability of the affected Functions.

In addition to the channel OPERABILITY guidance discussed above, the COT and CHANNEL CALIBRATION SRs specified on Table 3.3.2-1 for certain ESFAS Functions are modified by Notes (e) and (f) that specify additional Technical Specification requirements. The applicable Notes are specified directly on Table 3.3.2-1 next to the numerical SR designations for the affected RTS Functions. The additional Technical Specification requirements for these ESFAS Functions include OPERABILITY evaluations for setpoints found outside the as-found acceptance criteria band and the requirement to reset the setpoint to within the as-left tolerance of the nominal trip setpoint or a value that is more conservative than the nominal trip setpoint or declare the affected channel inoperable. These additional Technical Specification requirements are only applicable to the ESFAS Functions with the Notes modifying their COT and CHANNEL CALIBRATION SR numbers on Table 3.3.2-1.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

Provided for Information Only.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to ≤ 2200°F), and

2. Boration to ensure recovery and maintenance of SDM ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation,
- Reactor Trip,
- Turbine Trip,
- Feedwater Isolation,
- Start of auxiliary feedwater (AFW) pumps, and
- Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations,
- Trip of the turbine and reactor to limit power generation,
- Isolation of main feedwater (MFW) to limit secondary side mass losses,
- Start of AFW to ensure secondary side cooling capability, and
- Enabling ECCS suction switchover from the refueling water storage tank (RWST) ~~switchover~~to the containment sump on ~~low (Unit 1) and extreme low (Unit 2)~~ RWST Level Extreme Low ~~level~~ to ensure continued cooling via use of the containment sump.

No change. Page included for context only.

Provided for Information Only.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Safety Injection - Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals except for the Unit 1 automatic high head safety injection (HHSI) flow path isolation valves when LCO 3.4.12, "Overpressure Protection System," is applicable. Consistent with the requirements of LCO 3.4.12, in MODE 4 when any RCS cold leg temperature is ≤ the enable temperature specified in the PTLR, the Unit 1 automatic HHSI flow path must be isolated with power removed from the isolation valves. Therefore, when operating in the MODE 4 Applicability of LCO 3.4.12, the manual initiation of Unit 1 SI will require additional manual valve operation to establish an SI injection flow path.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one switch and the interconnecting wiring to the actuation logic cabinet. Each switch actuates both trains. This configuration does not allow testing at power.

b. Safety Injection - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

In the event an inadvertent SI is initiated, the block of the automatic actuation logic introduced by a reset of safety injection must be removed by resetting (closure) of the reactor trip breakers after the inadvertent initiation providing that all trip input signals have reset due to stable plant conditions. When the Automatic Actuation Logic is required OPERABLE and is blocked after an inadvertent SI, the affected train(s) of Automatic Actuation Logic are considered inoperable and the Technical Specification ACTIONS are applicable until the Automatic Actuation Logic is restored to OPERABLE status.

No change. Page included for context only.

Provided for Information Only.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4; however, only the actuation relays are required to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment, and
- Feed line break inside containment.

Containment Pressure - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic.

The high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure - High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve,
- SLB,
- A spectrum of rod cluster control assembly ejection accidents (rod ejection),
- Inadvertent opening of a pressurizer relief or safety valve,
- LOCAs, and
- SG Tube Rupture.

The Pressurizer Pressure - Low protection Function provides no input to any control functions. Pressurizer pressure control is accomplished by two separate channels independent of the pressurizer pressure protection channels used for ESFAS. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic.

The transmitters could experience adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. Safety Injection - Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB,
- Feed line break, and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low also provides input to steam generator level control; however, only three OPERABLE channels per steam line are provided. If a steam pressure sensor fails high or low, the steam generator level control system would eventually recover based upon the level input alone, assuming that a high level or low level trip setpoint is not reached. If the steam generator level setpoint is reached and protective action is required, a reactor trip (on low steam generator level) or turbine trip (on high steam generator level) occurs automatically. In this case, steam generator level is used to mitigate the event and not steam pressure. A single failure in a steam generator level channel could be assumed; however, the reactor trip would still occur on steam generator level. A second failure in another steam pressure transmitter would not preclude a trip from occurring on steam generator level. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

The Unit 1 transmitters will not experience adverse environmental conditions during a secondary side break. The Unit 2 transmitters are located where they may experience adverse environmental conditions during a secondary side break outside containment. However, for Unit 2, the safety analysis limit for the steam line break inside containment is more limiting than the safety analysis limit for the steam line break outside containment. As such, the Unit 2 Trip Setpoint is based on the more limiting result of the safety analysis for a steam line break inside containment which does not require an adverse environmental uncertainty. The magnitude of the difference between the inside and outside safety analysis limits is greater than or equal to the potential error that could result from an adverse environment. Therefore, the trip setpoints for both units only reflect steady state instrument uncertainties.

Provided for Information Onlv.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function is anticipatory in nature and has a lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic steam line isolation via Containment Pressure-Intermediate High High, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

2. Containment Spray Systems

Containment Spray provides five primary functions:

1. Lowers containment pressure and temperature after an HELB in containment,

2. Reduces the amount of radioactive iodine in the containment atmosphere,

3. Adjusts the pH of the water in the containment recirculation sump after a large break LOCA,

4. Mixes the containment atmosphere and minimizes the amount of hydrogen accumulation, and

5. Removes containment heat.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure,
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure,
- Minimize corrosion of the components and systems inside containment following a LOCA,
- Control subcompartment and general area hydrogen concentrations to less than 4% by volume, and

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Remove decay heat to ensure that the containment gas and sump water temperatures are within the containment liner and piping thermal stress limits.

The containment spray actuation signal starts the Quench Spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is drawn from the RWST by the Quench Spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. The Quench Spray pumps are manually stopped ~~after recirculation operation is established~~~~when the~~ following receipt of a low RWST level alarm.~~is empty.~~ The Recirculation Spray pumps are started automatically ~~after a specified time delay~~ and take suction from the containment sump to continue containment spray. ~~Containment~~ Recirculation spray is actuated manually or by Containment Pressure - High~~-~~High coincident with RWST Level Low.

a.(1)~~.~~ ~~Containment~~ Quench Spray - Manual Initiation

The operator can initiate ~~containment~~quench spray at any time from the control room by simultaneously actuating two containment spray actuation switches in the same train. Because an inadvertent actuation of quench~~containment~~ spray could have undesirable consequences, two switches must be actuated simultaneously to initiate quench~~containment~~ spray. There are two sets of two switches each in the control room. Simultaneously actuating the two switches in either set will actuate quench~~containment~~ spray in both trains in Unit 2 and one train in Unit 1. Two Manual Initiation switches in each train are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. ~~Note that~~ Manual Initiation of quench~~containment~~ spray also actuates Phase B containment isolation. Note that manual initiation of containment spray will initiate a recirculation spray pump start if an RWST Level Low signal is present. Alternatively, an operator can individually start each recirculation spray pump using the ~~bench~~ control board pump switches.

~~b~~a.(2)~~.~~ ~~Containment~~ Quench Spray - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment. Manual and automatic initiation of quench~~containment~~ spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not

Provided for Information Only.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a ~~containment~~ quench spray, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4; however, only the actuation relays are required to support ~~system level~~ manual initiation of quench spray. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

~~c~~a.(3)~~.~~ ~~Containment~~ Quench Spray - Containment Pressure - High High

This signal provides protection against a LOCA or an SLB inside containment. The transmitters will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is one of two Functions that require the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate the containment spray systems.~~, due to the consequences of an inadvertent actuation of containment spray.~~ Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

This Function uses four channels in a two-out-of-four logic configuration. Additional redundancy is warranted because this Function is energized to trip. Containment Pressure - High High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High High setpoints.

b.(1) Recirculation Spray - Automatic Actuation Logic

This LCO requires two trains to be OPERABLE. The trains consist of the actuation logic and associated master relays for this Function. The actuation logic consists of all circuitry housed within the actuation subsystems. The LCO for this Function does not include requirements for slave relay OPERABILITY. The SRs for this Function do not include a SLAVE RELAY TEST due to equipment safety concerns (inadvertent pump start) if such a test was performed at power. The verification of required slave relay OPERABILITY for this Function is included in LCO 3.6.7 Recirculation Spray System (SR 3.6.7.3.b). The Recirculation Spray System SR is an 18 month Surveillance that allows the required SLAVE RELAY TEST to be performed safely. Therefore, LCO 3.6.7 addresses the OPERABILITY of the slave relays for this Function.

b.(2) Recirculation Spray – RWST Level Low and Containment Pressure High-High

This LCO requires ~~two trains~~three channels to be OPERABLE.

A Level Low in the RWST coincident with a Containment Pressure High High signal automatically initiates recirculation spray. Recirculation spray is the primary method of heat removal from the containment environment following a LOCA. The RWST Level Low Allowable Value has both upper and lower limits. The lower limit is selected to ensure that containment temperatures remain within safety analysis limits and that adequate NPSH is available to the LHSI pumps. The upper limit ensures adequate NPSH to the recirculation spray pumps.

The RWST Level Low Function uses three RWST level transmitters in a two out of three coincident logic. These transmitters provide no control functions. The transmitters will not experience any adverse environmental conditions and, therefore, the trip setpoint reflects only steady state instrument uncertainties. The RWST level logic is configured in a de-energize to trip configuration.

The Containment Pressure High-High signal is described in Quench Spray, Containment Pressure High-High (item 2.a.3).

The RWST Level Low and Containment Pressure High High Functions must be OPERABLE in MODES 1, 2~~,~~ and 3~~, and 4~~ when there is a potential for a LOCA to occur, to ensure a continued supply of water for the recirculation spray pumps. These Functions are not required to be OPERABLE in MODES 4, 5 and 6 because there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High High setpoints.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW) and cooling water to the containment air recirculation fan cooling coils, and the Unit 1 containment instrument air, at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation relays. CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and motors. The cooling water to the containment air recirculation fan cooling coils is not isolated by a Phase A signal to allow continued containment cooling. The Unit 1 containment instrument air is not isolated by a Phase A signal to allow instrument air to be available to support valve operation inside containment (e.g., CCW valves). All process lines required to be isolated under accident conditions and not equipped with automatic isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4 (except when open under administrative controls).

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains.

The Phase B signal isolates CCW and cooling water to the containment air recirculation fan cooling coils and containment instrument air (for Unit 1 only). This occurs at a relatively high containment pressure that is indicative of a large break LOCA or an SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating these additional systems at the higher pressure does not pose a challenge to the containment boundary because the systems are closed loops inside containment. The systems are continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation

No change. Page included for context only.

Provided for Information Only.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

of Phase B isolation would be into containment. Therefore, the combination of system design and Phase B isolation ensures the systems are not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by Containment Pressure - High High, or manually, via the automatic actuation relays, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure - High High, a LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCW to the RCPs is, therefore, no longer necessary.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches in either set are actuated simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains in Unit 2 and one train in Unit 1.

a. <u>Containment Isolation - Phase A Isolation</u>

(1) <u>Phase A Isolation - Manual Initiation</u>

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains.

(2) <u>Phase A Isolation - Automatic Actuation Logic and Actuation Relays</u>

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4; however,

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

only the actuation relays are required to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation - Phase B Isolation

Phase B Containment Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels (the same channels that actuate Containment Spray, Function 2.c). The Containment Pressure actuation of Phase B Containment Isolation is energized to actuate in order to minimize the potential of spurious actuations that may damage the RCPs.

(1) Phase B Isolation - Manual Initiation

The manual Phase B Containment Isolation is accomplished by the manual Containment Spray switches described in Function 2.a.

(2) Phase B Isolation - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a Phase B containment isolation, actuation is simplified by the use of the manual actuation switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4; however, only the actuation relays are required to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation - Containment Pressure - High High

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. For Unit 2 which does not have steam line check valves, Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation - Manual Initiation (Unit 2 only)

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two switches per train in the control room and simultaneous actuation of both switches in a

Provided for Information Only.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The SRs for this Function do not include a SLAVE RELAY TEST due to equipment safety concerns if such a test was performed at power. The verification of required slave relay OPERABILITY for this Function is included in LCO 3.5.2, ECCS - Operating (SRs 3.5.2.5 and 3.5.2.6). These ECCS SRs are 18 month Surveillances that allow the required SLAVE RELAY TEST to be performed safely. Therefore, LCO 3.5.2 addresses the OPERABILITY of the slave relays for this Function.

b. Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level ~~- Low (Unit 1) and~~ Extreme Low ~~(Unit 2)~~ Coincident With Safety Injection

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A ~~low (Unit 1) and~~ Level ~~E~~extreme ~~L~~low ~~(Unit 2) level~~ in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The SI interlock is maintained by latching relays until reset manually. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability due to the energize to trip design of these channels.

The RWST ~~- low (Unit 1) and~~ Level ~~E~~extreme ~~L~~low ~~(Unit 2)~~ Allowable Value has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage. The upper limit is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The upper limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

The transmitters will not experience any adverse environmental conditions and, therefore, the trip setpoint reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST ~~low (Unit 1) and~~ Level ~~E~~extreme ~~L~~low ~~(Unit 2) level~~ signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. Although SI actuation may be manually reset after a 75 second delay, if P-4 is enabled, subsequent automatic SI initiation is blocked until P-4 is reset (RTBs closed). This Function allows operators to take manual control of SI systems after the initial phase of injection is complete without further automatic SI actuations taking place. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate MFW Regulating Valves with coincident low T_{avg},
- Prevent automatic reactuation of SI after a manual reset of SI, and
- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level - High High with low T_{avg}.

Provided for Information Only.

BASES

ACTIONS (continued)

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 5) that 4 hours is the average time required to perform channel surveillance.

<u>D.1, D.2.1, and D.2.2</u>

Condition D applies to:

- Containment Pressure - High,
- Pressurizer Pressure - Low,
- Steam Line Pressure - Low,
- Containment Pressure - Intermediate - High High,
- Steam Line Pressure - Negative Rate - High,
- SG Water level - Low Low, and
- SG Water level - High High (P-14)~~.~~
- <u>RWST Level Low.</u>

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

ACTIONS (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 5.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure - High High, and
- Containment Phase B Isolation Containment Pressure - High High.

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable

ACTIONS (continued)

condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 5.

<u>I.1 and I.2</u>

Condition I applies to the AFW pump start on trip of all MFW pumps.

The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 5.

<u>J.1, J.2.1, and J.2.2</u>

Condition J applies to:

- RWST Level ~~- Low (Unit 1) and~~ Extreme Low ~~(Unit 2)~~ Coincident with Safety Injection.

RWST Level ~~- Low (Unit 1) and~~ Extreme Low ~~(Unit 2)~~ Coincident With SI provides actuation of switchover to the containment sump. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies

No change. Page included for context only.

Provided for Information Only.

ACTIONS (continued)

the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed low). If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The 6 hours to place a channel in bypass and the total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference 5.

K.1, K.2.1, and K.2.2

Condition K applies to the P-11 and P-12 interlocks.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock.

Determination must be made within 1 hour and may be made by observation of the associated permissive annunciator window(s) (bistable status lights or computer checks). The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

Provided for Information Only.

SURVEILLANCE REQUIREMENTS (continued)

For the Unit 2 recirculation spray pumps 2RSS-P21C and 2RSS-P21D, the term "required developed head" refers to the value that is assumed in the Containment Integrity Safety Analysis for the recirculation 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 MOP in the 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. SRs are specified in the IST Program of the ASME Code. The ASME Code provides the activities and frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump, except 2RSS-P21C and 2RSS-P21D, starts on receipt of an actual or simulated SI signal. The Unit 2 recirculation spray pumps 2RSS-P21C and 2RSS-P21D start on a receipt of an actual or simulated ~~RWST Level Low and coincidence~~ Containment Pressure - High High and RWST Level Low signal or a coincidence RWST Level Extreme Low and SI signal. ~~following a time delay.~~

For the Automatic Switchover to the Containment Sump function of the ECCS, these Surveillances include a verification of the associated required slave relay operation. The Automatic Switchover to the Containment Sump, Function 7 in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," does not include a requirement to perform a SLAVE RELAY TEST due to equipment safety concerns if such a test was performed at power. Therefore, verification of the required slave relay OPERABILITY for the Automatic Switchover to the Containment Sump ESFAS function is included in these 18-month ECCS Surveillances. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls.

The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. However, this does not preclude performance of this Surveillance at power when it can be accomplished in

Provided for Information Only.

BASES

SURVEILLANCE REQUIREMENTS (continued)

a safe manner. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

Periodic inspections of the accessible regions of the containment sump suction inlet ~~and~~ strainers ensure that ~~it is~~ they are unrestricted, free of structural distress or abnormal corrosion, and stay~~s~~ in proper operating condition. Accessible regions of the sump strainers are those regions that can be visually examined without disassembling the strainer assembly or the grating and cover plates over the strainer assembly~~unit~~. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and on the need to have access to the location. However, this does not preclude performance of this Surveillance at power when it can be accomplished in a safe manner. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. UFSAR, Appendix 1A, "1971 AEC General Design Criteria Conformance, " (Unit 1) and UFSAR, Section 3.1, "Conformance with U.S. Nuclear Regulatory Commission General Design Criteria, " (Unit 2).
2. 10 CFR 50.46.
3. UFSAR, Section 14.3 (Unit 1) and UFSAR, Section 15.6.5 (Unit 2).
4. UFSAR, Section 14.3.4 (Unit 1) and UFSAR, Section 6.2.1 (Unit 2).
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

Provided for Information Only.

BASES

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. A main steam line break inside containment is not evaluated as the dose consequences are bounded by a main steam line break outside containment. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = ~~43.3~~43.1 psig (for Unit 1) and ~~44.9~~44.8 psig (for Unit 2) (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except during the first unit startup prior to entering MODE 4 after performing a required Containment Leakage Rate Testing Program leakage test. At this time the other applicable leakage limits specified in the Containment Leakage Rate Testing Program must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates for the containment air lock (LCO 3.6.2) are specified in the Containment Leakage Rate Testing Program and are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding the air lock limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

Provided for Information Only.

APPLICABLE SAFETY ANALYSES (continued)

of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 3), as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = ~~43.3~~43.1 psig (for Unit 1) and ~~44.9~~44.8 psig (for Unit 2) following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, 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.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. However, if the inner door is inoperable it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable

B 3.6 CONTAINMENT SYSTEMS

Provided for Information Only.

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Quench Spray System. In addition, the lower containment pressure limit provides assurance that sufficient net positive suction head exists for the pumps taking suction from the containment sump during the recirculation phase of operation after a LOCA.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes. The worst case LOCA results in a higher containment pressure than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure assumed in the containment analysis was 14.2 psia. This resulted in a maximum peak pressure from a LOCA of ~~43.3~~ 43.1psig (Unit 1) and ~~44.9~~44.8 psig (Unit 2). The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, ~~43.3~~43.1 psig (Unit 1) and ~~44.9~~44.8 psig (Unit 2), does not exceed the containment design pressure, 45 psig.

The containment was also designed for an internal pressure of 8.0 psia. The inadvertent actuation of the Quench Spray System was evaluated to determine the resulting reduction in containment pressure. The initial pressure condition used in this evaluation was 12.8 psia. This resulted in a minimum pressure inside containment of ~~11.44~~11.38 psia, which is within the containment design capability.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

No change. Page included for context only.

BASES

Provided for Information Only.

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Quench and Recirculation Spray systems during post accident conditions is dependent upon the energy released to the containment due to the event. Higher initial temperature results in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Too low a containment temperature would adversely impact the small break LOCA safety analysis assumptions regarding the automatic actuation of Phase B containment isolation on containment high-high pressure. As such, operation with containment temperature outside the LCO limits violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses and is important in establishing environmental qualification (EQ) requirements to assure the required equipment inside containment performs as designed during and after a DBA. The upper limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1). The lower containment temperature limit ensures that Containment Isolation Phase B will be actuated by the Containment Pressure - High High setpoint consistent with the assumptions of the small break LOCA analysis.

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The SLB resulted in the maximum calculated peak containment temperature and containment liner temperature. The Unit 1 SLB that resulted in the peak containment temperature occurred at

APPLICABLE SAFETY ANALYSES (continued)

100% RTP, with the worst case single failure of a main steam check valve. The Unit 1 SLB that resulted in the peak containment liner temperature occurred at 30% RTP, with the worst case single failure of a main steam check valve. The Unit 2 SLB that resulted in the peak containment ~~and containment liner~~ temperature occurred at ~~0~~100% RTP, with the worst case single failure of a main steam isolation valve. The Unit 2 SLB that resulted in the peak containment liner temperature occurred at 0% RTP, with the worst case single failure of a main steam isolation valve.

The initial upper containment average air temperature assumed in the design basis analyses (Ref. 1) is ~~105~~108°F. This resulted in a maximum containment air temperature of ~~354.4~~355.9°F (for Unit 1) and ~~335.1~~345.6°F (for Unit 2) and a maximum containment liner temperature of ~~254.1~~257.9°F (for Unit 1) and ~~247.7~~248.7°F (for Unit 2). The design temperature of the containment liner is 280°F.

The containment air temperatures resulting from DBAs are used to establish EQ requirements (Ref. 2) for equipment inside containment. The EQ requirements provide assurance the equipment inside containment required to function during and after a DBA performs as designed during the adverse environmental conditions resulting from a DBA. Air temperature profiles (containment air temperature vs time) are calculated for each DBA to establish EQ design requirements for the equipment inside containment. The equipment inside containment required to function during and after a DBA is confirmed to be capable of performing its design function under the applicable EQ requirement (i.e., air temperature profile). Maintaining the initial containment air temperature within the required limits preserves the initial conditions assumed in the accident analyses which limits the containment air temperature and pressure resulting from various DBAs. Limiting the containment air temperature and pressure that result from various DBAs ensures the equipment inside containment will continue to perform as designed during and after a DBA. Therefore, it is concluded that the calculated transient containment air temperature resulting from various DBAs, including the most limiting temperature from a SLB, are acceptable.

The upper temperature limit is also used in the depressurization evaluation to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Quench Spray System (Ref. 3).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature limit is used in this analysis to ensure that in the event of an accident the design containment internal pressure will not be exceeded.

B 3.6 CONTAINMENT SYSTEMS

No change. Page included for context only.

B 3.6.6 Quench Spray (QS) System

Provided for Information Only.

BASES

BACKGROUND

The QS System is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS System, operating in conjunction with the Recirculation Spray (RS) System, is designed to cool and depressurize the containment structure to less than 50% of the peak calculated containment pressure within 24 hours following a Design Basis Accident (DBA). Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The QS System consists of two separate trains of adequate capacity, each capable of meeting the design bases. Each train includes a spray pump, spray headers, nozzles, valves, and piping. The two Unit 2 containment spray ring headers are shared by both QS System trains. Each train is powered from a separate Engineered Safety Features (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the QS System.

The QS System is actuated either automatically by a Containment High-High pressure signal or manually. The QS System provides a spray of cold borated water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Each train of the QS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The Unit 1 QS System also provides flow to the containment sump to improve the net positive suction head available to the RS System pumps.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the spray. The Unit 2 Spray Additive System also injects NaOH solution directly into the containment sump. The resulting alkaline pH of the spray and Unit 2 containment sump enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The NaOH added to the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The QS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS System and RS System provides the required heat removal capability to limit post accident conditions to

Provided for Information Only.

BACKGROUND (continued)

less than the containment design values and depressurize the containment structure to less than 50% of the peak calculated containment pressure within 24 hours following a DBA.

The QS and RS Systems limit the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, with respect to the worst case single active failure. The appropriate single failure is assumed in the safety analysis. However, the maximum calculated peak containment pressure results from a LOCA postulated to occur in the RCS hot leg. The calculated peak containment pressure from this location occurs during the blowdown phase, prior to the actuation of any safety related equipment, consequently there is no single failure assumed in this analysis. The SLB resulted in the maximum calculated peak containment temperature and containment liner temperature. The Unit 1 SLB that resulted in the peak containment temperature occurred at 100% RTP, with the worst case single failure of a main steam check valve. The Unit 1 SLB that resulted in the peak containment liner temperature occurred at 30% RTP, with the worst case single failure of a main steam check valve. The Unit 2 SLB that resulted in the peak containment temperature occurred at 100% RTP and containment liner temperature occurred at 0% RTP, with the worst case single failure of a main steam isolation valve.

During normal operation, the containment internal pressure is maintained within the limits of LCO 3.6.4, "Containment Pressure." Maintaining containment pressure within the required limits during operation ensures the capability to depressurize the containment to less than 50% of the peak calculated containment pressure within 24 hours after a DBA.

The DBA analyses (Ref. 1) show that the maximum peak containment pressure of ~~43.3~~43.1 psig (Unit 1) and ~~44.9~~44.8 psig (Unit 2) results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature of ~~354.4~~355.9°F (Unit 1) and ~~335.1~~345.6°F (Unit 2) and the maximum containment liner temperature of ~~254.1~~257.9°F (Unit 1) and ~~247.7~~248.7°F (Unit 2) results from the SLB analysis. The containment liner design temperature is 280°F. The containment air temperatures resulting from DBAs are used to establish

B 3.6 CONTAINMENT SYSTEMS

Provided for Information Only.

B 3.6.7 Recirculation Spray (RS) System

BASES

BACKGROUND

The RS System, operating in conjunction with the Quench Spray (QS) System, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to less than 50% of the peak calculated containment pressure within 24 hours following a Design Basis Accident (DBA). The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a DBA.

The RS System consists of two separate trains of adequate capacity, each capable of meeting the design and accident analysis bases.

Unit 1

The Unit 1 Recirculation Spray System consists of four 50 percent capacity subsystems (2 per train). Each subsystem is composed of a spray pump, associated heat exchanger and flow path. Two of the recirculation spray pumps are located outside containment (RS-P-2A and RS-P-2B) and two pumps 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.

Unit 2

The Unit 2 Recirculation Spray System consists of four 50 percent capacity subsystems (2 per train). Each subsystem is composed of a spray pump, associated heat exchanger and flow path. All recirculation spray pumps 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 Level Eextreme Llow- ~~level~~setpoint, the C and D subsystems are automatically switched to the cold leg recirculation mode of Emergency Core Cooling System (ECCS) operation.

Provided for Information Only.

BACKGROUND (continued)

Each train of the RS System provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal.

The RS System provides a spray of subcooled water into the upper regions of containment to reduce the containment pressure and temperature during a DBA. Upon receipt of a coincident High-High C~~c~~ontainment P~~p~~ressure signal (Containment Isolation Phase B (CIB)) and a RWST Level Low signal, the Unit 1 RS-P-1A and RS-P-~~2~~1B pumps immediately start.~~ after a 210 ± 5 second time delay and t~~The Unit 1 RS-P-2A and RS-P-~~1~~2B pumps start after a ~~225 ± 5~~15 second time delay for emergency generator loading considerations. All the Unit 2 RS pumps start immediately following receipt of the actuation signals.~~after a 628 ± 5 second time delay.~~ The RS pumps take suction from the containment sump and discharge through their respective spray coolers to the spray headers and into the containment atmosphere. Heat is transferred from the containment sump water to river/service water in the spray coolers.

The Spray Additive System injects a sodium hydroxide (NaOH) solution into the suction of the QS System pumps. The Unit 2 Spray Additive System also injects NaOH solution directly into the containment sump. The NaOH added to the QS System spray and Unit 2 containment sump ensures an alkaline pH for the solution recirculated in the containment sump. The resulting alkaline pH of the RS spray (pumped from the sump) enhances the ability of the spray to scavenge iodine fission products from the containment atmosphere. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The RS System is a containment ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained. Operation of the QS and RS systems provides the required heat removal capability to limit post accident conditions to less than the containment design values and depressurize the containment structure to less than 50% of the peak calculated containment pressure within 24 hours following a DBA.

The RS System limits the temperature and pressure that could be expected following a DBA and ensures that containment leakage is maintained consistent with the accident analysis.

Provided for Information Only.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients; DBAs are assumed not to occur simultaneously or consecutively. The postulated DBAs are analyzed assuming the worst case single active failure. The appropriate single failure is assumed in the safety analysis. However, the maximum calculated peak containment pressure results from a LOCA postulated to occur in the RCS hot leg. The calculated peak containment pressure from this location occurs during the blowdown phase, prior to the actuation of any safety related equipment, consequently there is no single failure assumed in this analysis. The SLB resulted in the maximum calculated peak containment temperature and containment liner temperature. The Unit 1 SLB that resulted in the peak containment temperature occurred at 100% RTP, with the worst case single failure of a main steam check valve. The Unit 1 SLB that resulted in the peak containment liner temperature occurred at 30% RTP, with the worst case single failure of a main steam check valve. The Unit 2 SLB that resulted in the peak containment ~~and containment liner~~ temperature occurred at ~~0~~100% RTP, with the worst case single failure of a main steam isolation valve (Ref. 1). The Unit 2 SLB that resulted in the peak containment liner temperature occurred at 0% RTP, with the worst case single failure of a main steam isolation valve (Ref. 1).

The peak containment pressure following a high energy line break is affected by the initial total pressure and temperature of the containment atmosphere. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure.

During normal operation, the containment internal pressure is maintained within the limits of LCO 3.6.4, "Containment Pressure." Maintaining containment pressure within the required limits during operation ensures the capability to depressurize the containment to less than 50% of the peak calculated containment pressure within 24 hours after a DBA. This capability and the variation of containment pressure are functions of river/service water temperature, RWST water temperature, and the containment air temperature.

The DBA analyses show that the maximum peak containment pressure of ~~43.3~~43.1 psig (Unit 1) and ~~44.9~~44.8 psig (Unit 2) results from the LOCA analysis and is calculated to be less than the containment design pressure. The maximum containment atmosphere temperature of ~~354.4~~355.9 °F (Unit 1) and ~~335.1~~346.6 °F (Unit 2) and the maximum containment liner temperature of ~~254.1~~257.9 °F (Unit 1) and ~~247.7~~248.7 °F (Unit 2) result from the SLB analysis. The containment liner design temperature is 280°F. The containment air temperatures resulting from DBAs are used to establish equipment qualification (EQ) requirements (Ref. 2) for equipment inside containment.

Provided for Information Only.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EQ requirements provide assurance the equipment inside containment required to function during and after a DBA performs as designed during the adverse environmental conditions resulting from a DBA. Air temperature profiles (containment air temperature vs time) are calculated for each DBA to establish EQ design requirements for the equipment inside containment. The equipment inside containment required to function during and after a DBA is confirmed to be capable of performing its design function under the applicable EQ requirement (i.e., air temperature profile). Therefore, it is concluded that the calculated transient containment atmosphere temperatures resulting from various DBAs, including the most limiting temperature from a SLB, are acceptable. The RS System is not credited in the SLB containment analysis.

The RS System actuation model from the containment analysis is based upon a response time ~~associated with exceeding the High-High containment pressure signal and RWST Level Low signal setpoint~~between receipt of the RWST Level Low signal in coincidence with the Containment Pressure High-High to achieving full flow through the RS System spray nozzles. A delay in response time initiation provides conservative analyses of peak calculated containment temperature and pressure. The RS System maximum time from coincidence of Containment Pressure High-High~~CIB~~ and RWST Level Low to the start of effective RS spray ~~(which includes the pump start delay time)~~ is ~~300~~65 seconds for Unit 1 and ~~715~~ 77 seconds for Unit 2.

In the case of the Unit 2 RS System, the containment safety analysis models the operation of the system consistent with the system design. The Unit 2 analysis models the RS subsystems starting in the spray mode of operation. When the unit is shifted to the ECCS recirculation mode of operation the containment analysis models a reduction in recirculation spray flow to account for the Unit 2 RS subsystems used for the ECCS low head recirculation function.

For certain aspects of accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

The RS System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.3

These SRs ensure that each automatic valve actuates and that the RS System pumps start upon receipt of an actual or simulated ~~High-High containment pressure and~~ coincident with a Containment Pressure High-High/RWST Level Low signal.~~following a time delay. Start delay times are also verified for the RS System pumps.~~ However, t~~T~~he Unit 1 RS-P-~~1~~2A and RS-P-2B pumps start after an additional delay of ~~210 ± 5~~15 second time delay for emergency diesel generator loading considerations. The start delay time is also verified for the RS System pumps.~~The Unit 1 RS-P-2A and RS-P-1B pumps start after a 225 ± 5 second time delay. The Unit 2 pumps start after a 628 ± 5 second time delay.~~

For the RS function of the Containment Spray System, this Surveillance includes a verification of the associated required slave relay operation. Recirculation Spray – Automatic Actuation, Function 2.b.1 in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," does not include a requirement to perform a SLAVE RELAY TEST due to equipment safety concerns if such a test was performed at power. Therefore, verification of the required slave relay OPERABILITY for the Recirculation Spray-Automatic Actuation, Function 2.b.1 in LCO 3.3.2 is included in this 18-month Surveillance.

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

SR 3.6.7.4

This SR is performed following maintenance when the potential for nozzle blockage has been determined to exist by an engineering evaluation. The required evaluation will also specify an appropriate test method for determining the spray ring OPERABILITY. Due to the passive design of the spray rings and their normally dry state, a test following maintenance that results in the potential for nozzle blockage is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. UFSAR, Chapter 14 (Unit 1), and UFSAR, Section 6.2 (Unit 2).
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME code for Operation and Maintenance of Nuclear Power Plants.

Enclosure Attachment C-1

Beaver Valley Power Station, Unit No. 1
Proposed Licensing Requirements Manual Changes

License Amendment Request 334 (Unit 1)

Licensing Requirements Manual changes are provided for Information Only

The following is a list of the affected pages:

Page
3.2-3
3.9-4

Provided for Information Only.

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)	≤ 17.0[10]/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	≤ ~~81.5~~85.5[9]
	b.	Containment Isolation-Phase "B"	Not Applicable
	c.	Control Room Ventilation Isolation	≤ 22.0[8]/77.0[7]

Provided for Information Only.

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	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 Extreme-~~-~~Low	14~~'~~ feet 1/2~~"~~ inches***
2. CONTAINMENT SPRAY SYSTEMS	
a. Quench Spray	
~~a~~1. Manual	Not Applicable
~~b~~2. Automatic Actuation Logic	Not Applicable
~~c~~3. Containment Pressure--High-High	11.1 psig**
b. Recirculation Spray	
1. Automatic Actuation Logic and Actuaton Relays	Not Applicable
2. Refueling Water Storage Tank (RWST) Level Low Coincident with Containment Pressure High-High	27 feet 7.5 inches*** 11.1 psig**

* The Unit 1 Setpoint Methodology used to establish the Engineered Safety Feature Actuation System Setpoints is defined in WCAP 11419.

** The predefined as-found acceptance band, and the as-left setpoint tolerance is ± 0.33 psig.

*** The predefined as-found acceptance band, and the as-left setpoint tolerance is ± 3.5 inches.

****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.

Enclosure Attachment C-2

Beaver Valley Power Station, Unit No. 2
Proposed Licensing Requirements Manual Changes

License Amendment Request 205 (Unit 2)

Licensing Requirements Manual changes are provided for Information Only

The following is a list of the affected pages:

Page
3.10-4

TABLE 3.10-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	NOMINAL TRIP SETPOINT[1]
1.	SAFETY INJECTION AND FEEDWATER ISOLATION	
	a. Manual Initiation	N.A.
	b. Automatic Actuation Logic and Actuation Relays	N.A.
	c. Containment Pressure - High	5.0 psig[3]
	d. Pressurizer Pressure - Low	1856 psig
	e. Steamline Pressure - Low	500 psig[2]
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	31 feet 9 inches[4]
2.	CONTAINMENT SPRAY SYSTEMS	
	a. Quench Spray	
	~~a~~1. Manual Initiation	N.A.
	~~b~~2. Automatic Actuation Logic and Actuation Relays	N.A.
	~~c~~3. Containment Pressure --High-High	11.1 psig[3]
	b. Recirculation Spray	
	1. Automatic Actuation Logic and Actuaton Relays	Not Applicable
	2. Refueling Water Storage Tank (RWST) Level Low Coincident with Containment Pressure High-High	32 feet 9 inches[4] 11.1 psig[3]

(1) The Unit 2 Setpoint Methodology used to establish the Engineered Safety Feature Actuation System Setpoints is defined in WCAP 11366.

(2) 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.

(3) The predefined as-found acceptance band, and the as-left tolerance band is ± 0.3 psig.

(4) The predefined as-found acceptance band, and the as-left tolerance band is ± 1".

Enclosure Attachment D

Beaver Valley Power Station, Unit Nos. 1 and 2

License Amendment Requests 334 (Unit 1) and 205 (Unit 2)

Containment and Net Positive Suction Head Analyses

Analysis Methodology

Beaver Valley Power Station (BVPS) recently converted to an atmospheric containment design basis. This features normal operation at slightly sub-atmospheric pressure conditions. This change to the BVPS operating license was approved by the NRC in Amendments 271 and 153 for Units 1 and 2, respectively. For purposes of the analysis documented in this attachment, and to be consistent with plant terminology, a Containment Pressure High-High signal is equivalent to a containment isolation phase B (CIB) signal. The methodology for performing these analyses includes the use of the MAAP-DBA code and is consistent with the current licensing basis. Details of this methodology are discussed in Reference D-1. Acceptance of this methodology for BVPS is documented in Reference D-2. The following discussion highlights key elements of the methodology:

- The analysis includes a wide range of large and small break loss of coolant accidents and Main Steam Line Breaks (MSLB). Sensitivity studies were also re-done to establish the limiting direction of bias for key input parameters based on attributes of interest (peak pressure, peak temperature, NPSH, etc.).

- For calculating containment peak pressure, temperature profiles, and maximum liner temperatures for Large Break loss of coolant accident (LOCA) and MSLB events, a single node containment model is used. This model employs the use of conservative heat transfer correlations. The Tagami correlation is used for LOCA events and the Uchida correlation is used for MSLB events.

- A multiple node containment model is used for calculations where water holdup in containment compartments is important in establishing conservative results. This includes Engineered Safety Feature (ESF) pump NPSH calculations, sump temperature calculations, and minimum sump inventory calculations. The nodalization is consistent with the analysis described in Reference D-1 as shown in Figures D-1 and D-2.

- The mass and energy release information used in the analyses is consistent with the approach used in the current licensing basis calculations. For Large Break LOCA and MSLB events, mass and energy release rates are calculated using NRC approved methodologies documented in References D-3 and D-4. The mass and energy release inputs for these analyses are not affected by the modifications proposed in this submittal and have not changed from the current licensing basis values. For Small Break LOCA, MAAP-DBA is used to calculate the mass and energy release rates. This is also consistent with the current licensing basis methods.

Analysis Acceptance Criteria

The current design basis requires that the Containment Integrity and ESF pump NPSH analyses meet the following criteria:

- Containment peak pressure following any accident must remain less than design
- Containment temperatures following any accident must remain less than the equipment capabilities for any equipment credited for mitigation
- Containment peak liner temperature must remain less than the design temperature
- Containment Pressure must be reduced to less than 50% of the peak pressure within 24 hours following the initiation of the accident
- The NPSH available for ESF pumps which draw from the containment sump must meet or exceed the required NPSH
- Containment minimum sump inventory (level) must meet the replacement screen design requirements
- Containment sump temperature must be below design temperature for piping systems which carry this fluid

Analysis Input Changes

Modifications to analysis inputs were required to be implemented to accurately model the changes proposed in this submittal. Some additional model input changes were also implemented to either enhance the accuracy of the model or recover margin in specific areas. The changes are described below:

- The signal to start the RSS pumps was changed from a fixed time delay following Containment Pressure High-High (equivalent to CIB), to a start signal based on RWST Level Low coincident with a Containment Pressure High-High (equivalent to CIB) signal. The nominal level setpoints are 27 feet 7.5 inches for Unit 1 and 32 feet 9 inches for Unit 2. Safety analysis values used in MAAP-DBA consider instrument uncertainties in the direction which is conservative for the particular analysis. An explicit calculation of the RWST drawdown is performed integral within MAAP-DBA based on the flows being discharged from the tank.

- Minimum system performance curves input into the MAAP-DBA model for the Quench Spray (QS) and Recirculation Spray (RS) were lowered to allow additional margin for pump degradation and to account for uncertainty in instrumentation during surveillance testing of the pumps.

- The maximum containment initial temperature was increased from 105°F to 108°F. The purpose of raising this safety analysis limit is to allow for an increase in the containment operating temperature band.

- An improvement was made in the modeling of the direct QS injection flow to the RS pumps suctions at Unit 1. The improvement provides for a more accurate distribution between the inside RS (IRS) pump and the outside RS (ORS) pump. In the current analysis, the direct QS injection flows are averaged between the IRS and ORS pumps.

- The additional mass and surface area of the new containment strainer was added to the inventory of containment heat sinks. The current model did not include the existing containment screens since they do not contain a significant mass of metal. The new strainers contain approximately 40,000 pounds of stainless steel.

- The RS and Low Head Safety Injection (LHSI) pump suction friction loss values were adjusted to remove the existing sump screen contribution. This allows the resulting calculated NPSH margins (NPSH available – NPSH required) to be compared to the head loss for the new strainers.

<u>Analysis Results</u>

The peak containment pressures following a LOCA are shown in Tables D-1 and D-2 for Unit 1 and Unit 2. All results meet the containment design pressure limit of 45 psig. The pressure results show a small (0.1 to 0.2 psig) reduction from the current licensing basis analysis. The LOCA peak pressure is not affected by the change to the RSS start signal or timing since the peak occurs prior to operation of any containment spray system. The addition of the new containment strainer metal heat sink provides a small benefit. The tables also show the time required for containment pressure to be reduced to less than 50% of the peak value. All times meet the acceptance criteria of 24 hours. Figures D-3 and D-4 show the transient pressure profiles following a LOCA.

Tables D-3 and D-4 show the containment peak pressure results for a MSLB. For Unit 1, 4.6 square foot breaks are included; however, since the flow restrictor nozzle is now integral with the new steam generators installed early in 2006, these break sizes are no longer applicable. The 1.4 square foot break provides the limiting peak pressure result. All peak pressure results for both Unit 1 and Unit 2 meet the containment design pressure limit of 45 psig and are bounded by the LOCA results. The MSLB analysis does not credit operation of the RSS pumps and is therefore unaffected by the change in the start signal. The peak pressure increases slightly (0.1 psig) due to a decrease in the QS system flow. Input changes to RS and QS performance for this analysis were discussed above.

The containment temperature profiles for the limiting LOCA and MSLB cases are shown in Figures D-5 and D-6. The containment temperature profiles are used as input into the evaluations of equipment qualification. The highest peak temperature results from a MSLB. For both Unit 1 and Unit 2, the peak temperature increased by less than 1°F above the current Equipment Qualification (EQ) profiles. This is primarily due to an increase of 3°F in the initial containment temperature assumption. The peak temperature for the limiting cases occurs prior to operation of any containment sprays. For the longer term post accident temperature, the LOCA results are limiting. The results showed an increase of less than 2°F above the current EQ profiles for a short period of time. The maximum calculated containment liner temperatures are shown in Tables D-5 and D-6. All containment liner temperature results met the acceptance limit of 280°F.

Containment sump water temperatures are also calculated and used as input into piping structural analyses. Tables D-7 and D-8 show the results of the sump temperature calculations at various points where flow paths become activated. The maximum sump temperatures at RSS pump start generally decreased due to the additional sump volume from the QS system during the extended delay in RSS pump start time. The sump temperatures at the switchover to cold leg recirculation increase by approximately 40°F due to the increased RSS pump start delay. Since the time between RSS pump start and switchover is reduced, less heat can be removed from the containment sump water during this period. The switchover to hot leg recirculation occurs much later (6 hours) and therefore the increase in sump temperature is much smaller (approximately 2°F). These changes in fluid temperatures will be used to evaluate the structural adequacy of the piping and support systems.

Tables D-9 and D-10 show the limiting NPSH results for the RSS and LHSI (Unit 1 only) pumps for Unit 1 and Unit 2, respectively. The available NPSH transient results for the most limiting cases are shown in Figures D-7 and D-8. Consistent with the current licensing basis, Unit 1 continues to credit containment overpressure for calculating available NPSH. At Unit 2, the analysis assumes that the sump vapor pressure is equal to containment pressure consistent with the Standard Review Plan Section 6.2.2 guidance. This is also consistent with the current licensing basis.

The Unit 1 results show a decrease in the available NPSH for the Inside Recirculation Spray (IRS) pumps and an increase for the Outside Recirculation Spray (ORS) pumps. This is primarily due to the modeling change for the direct QS injection flow as discussed above in the input changes section. The minimum NSPH margin for the RSS pumps is 4.9 feet. The Unit 1 LHSI pump available NPSH decreases when compared to the current analysis. This is due to the higher sump temperature (and vapor pressure) at switchover to cold leg recirculation. The minimum NPSH margin for the LHSI pump is 5.4 feet.

For the Unit 2 RSS pumps, the available NPSH increases when compared to the current analysis. This is due to the increase in sump level when the RSS pumps start. The NSPH margin for these pumps is 5.0 feet.

Figures D-9 and D-10 show the limiting minimum sump water level profiles for Unit 1 and Unit 2. This is an important parameter which is used to establish the design of the new strainers to ensure submergence requirements can be met. At Unit 1, the minimum water level at RSS pump start is approximately 4.0 feet above the bottom of the sump. At Unit 2, the minimum water level is approximately 6.7 feet above the bottom of the sump.

To enhance communications, the following acronyms used in the tables and figures are defined.

DEPS = Double-ended Pump Suction

DEHL = Double-ended Hot Leg

DER = Double-ended Rupture

References

D-1 FENOC letter L-04-073, "License Amendment Request Nos. 317 and 190," dated June 6, 20042

D-2 NRC letter dated February 6, 2006, "BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 (BVPS-1 AND 2) – ISSUANCE OF AMENDMENTS RE: CONTAINMENT CONVERSION FROM SUBATMOSPHERIC TO ATMOSPHERIC OPERATING CONDITIONS (TAC NOS. MC3394 AND MC3395," Amendments 271 and 153

D-3 WCAP-10325-P-A (Proprietary) and WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design – March 1979 Version," May 1983.

D-4 WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture", September1976

Table D-1 MAAP-DBA Containment Peak Pressure Results for a Design Basis Large Break LOCA for Unit 1				
Description	Power Level, %	Single Failure	Peak Pressure (psig) (1)	Time to reach <50% peak (hr)
6L-DEPS MIN SI	100.6	DG	41.9	0.8
7L-DEPS MAX SI	100.6	CIB	41.9	0.7
8L-DEHL	100.6	None	43.1	0.4

Single Failures – Failed Equipment
CIB One train each, QSS, RSS
DG One train each, SI, QSS, RSS
(1) Gauge pressure is referenced to 14.3 psi atmospheric pressure

Table D-2 MAAP-DBA Containment Peak Pressure Results for a Design Basis Large Break LOCA for Unit 2				
Description	Power Level, %	Single Failure	Peak Pressure (psig) (1)	Time to reach <50% peak (hr)
1L-DEPS MIN SI	100.6	DG	42.3	0.9
2L-DEPS MAX SI	100.6	CIB	42.3	0.7
3L-DEHL	100.6	None	44.8	0.7

Single Failures – Failed Equipment
CIB One train each, QSS, RSS
DG One train each, SI, QSS, RSS
(1) Gauge pressure is referenced to 14.3 psi atmospheric pressure

Table D-3 MAAP-DBA Peak Pressure Results for a Design Basis Main Steam Line Break for Unit 1			
Description	Power Level,%	SingleFailure*	Peak Pressure (psig)
1M-4.6 ft2 DER	100.6	MSCV	37.7
2M-4.6 ft2 DER	100.6	CIB/MFIV	31.2
3M-1.4 ft2 DER	100.6	MSCV	40.7
4M-1.4 ft2 DER	100.6	CIB/MFIV	33.2
5M-0.725 ft2 Split	100.6	MSCV	33.8
6M-0.725 ft2 Split	100.6	CIB/MFIV	32.7
7M-4.6 ft2 DER	70	MSCV	38.5
8M-4.6 ft2 DER	70	CIB/MFIV	32.4
9M-1.4 ft2 DER	70	MSCV	41.0
10M-1.4 ft2 DER	70	CIB/MFIV	33.9
11M-0.710 ft2 Split	70	MSCV	36.5
12M-0.710 ft2 Split	70	CIB/MFIV	34.8
13M-4.6 ft2 DER	30	MSCV	39.0
14M-4.6 ft2 DER	30	CIB/MFIV	32.8
15M-1.4 ft2 DER	30	MSCV	42.6
16M-1.4 ft2 DER	30	CIB/MFIV	35.2
17M-0.424 ft2 Split	30	MSCV	33.9
18M-0.424 ft2 Split	30	CIB/MFIV	34.3
19M-4.6 ft2 DER	0	MSCV	36.1
20M-4.6 ft2 DER	0	CIB/MFIV30	32.0
21M-1.4 ft2 DER	0	MSCV	41.4
22M-1.4 ft2 DER	0	MFIV30	33.9
22M1-1.4 ft2 DER	0	CIB	33.8
23M-0.209 ft2 Split	0	MSCV/MFIV30	26.5
24M-0.209 ft2 Split	0	CIB/MFIV30	30.5

Single Failures – Failed Equipment

CIB	One train QSS (RSS is conservatively not credited for MSLB evaluations)
DG	One train each, SI, QSS
MSCV	Main steam line check valve
MFIV	One main feedwater isolation valve
MFIV30	Main feedwater isolation valve (isolation time limited by FW bypass closure time of 30 seconds)

* Some of these cases assumed two active failures, one for M&E release and the other for containment response. This is a conservatism that helps control the number of cases in the run matrix. The M&Es for the double failure cases were not significantly different than if only a single failure had been assumed.

Table D-4
MAAP-DBA Peak Pressure Results for a Design Basis Main Steam Line Break for Unit 2

Description	Power Level,%	SingleFailure*	Peak Pressure (psig)
1M-1.069 ft2 DER	100.6	MSIV	36.9
2M-1.069 ft2 DER	100.6	MFIV/CIB	36.7
3M-0.753 ft2 Split	100.6	CIB	31.7
4M-0.753 ft2 Split	100.6	MSIV	32.4
5M-0.753 ft2 Split	100.6	MFIV	31.6
6M-1.069 ft2 DER	70	MSIV	37.1
7M-1.069 ft2 DER	70	MFIV/CIB	37.0
8M-0.757 ft2 Split	70	CIB	32.9
9M-0.757 ft2 Split	70	MSIV	33.7
10M-0.757 ft2 Split	70	MFIV	32.6
11M-1.069 ft2 DER	30	MSIV	39.4
12M-1.069 ft2 DER	30	MFIV/CIB	38.7
13M-0.756 ft2 Split	30	CIB	35.6
14M-0.756 ft2 Split	30	MSIV	36.4
15M-0.756 ft2 Split	30	MFIV	34.7
16M-1.069 ft2 DER	0	MSIV	38.1
17M-1.069 ft2 DER	0	MFIV/CIB	36.9
18M-0.608 ft2 Split	0	CIB	33.7
19M-0.608 ft2 Split	0	MSIV	32.1
20M-0.608 ft2 Split	0	MFIV	31.3

Single Failures – Failed Equipment

CIB	One train QSS (RSS is conservatively not credited for MSLB evaluations)
DG	One train each, SI, QSS
MSIV	One main steam isolation valve
MFIV	One main feedwater isolation valve

* Some of these cases assumed two active failures, one for M&E release and the other for containment response. This is a conservatism that helps control the number of cases in the run matrix. The M&Es for the double failure cases were not significantly different than if only a single failure had been assumed.

Table D-5 MAAP-DBA Peak Liner Temperatures for Unit 1			
LOCA		MSLB	
Sequence	Peak Liner Temperature °F	Sequence	Peak Liner Temperature °F
Case6L_Tagami_MLT	236.2	Case3M_Uchida_MLT_NRS	249.3
Case7L_Tagami_MLT	236.2	Case6M_Uchida_MLT_NRS	235.2
		Case9M_Uchida_MLT_NRS	252.8
		Case12M_Uchida_MLT_NRS	242.7
		Case15M_Uchida_MLT_NRS	257.9
		Case18M_Uchida_MLT_NRS	243.4
		Case22M1_Uchida_MLT_NRS	238.4
		Case24M_Uchida_MLT_NRS	240.6

Table D-6 MAAP-DBA Peak Liner Temperatures for Unit 2			
LOCA		MSLB	
Sequence	Peak Liner Temperature °F	Sequence	Peak Liner Temperature °F
Case1L_Tagami_MLT	236.4	Case2M_Uchida_MLT_NRS	241.5
Case2L_Tagami_MLT	236.5	Case3M_Uchida_MLT_NRS	233.1
		Case7M_Uchida_MLT_NRS	243.6
		Case8M_Uchida_MLT_NRS	237.2
		Case12M1_Uchida_MLT_NRS	247.1
		Case13M_Uchida_MLT_NRS	242.3
		Case16M_Uchida_MLT_NRS	248.7
		Case17M_Uchida_MLT_NRS	246.6
		Case18M_Uchida_MLT_NRS	244.5

Table D-7
MAAP-DBA ECCS and Recirculation Spray Piping Temperatures for Unit 1.

Case	Prior to Recirculation Spray System Start	Prior to Cold Leg Recirculation Transition	After Cold Leg Recirculation Transition	After Hot Leg Recirculation Transition(1)
6L	234°F	226°F	206.4°F	149°F
7L	234°F	225°F	198.5°F	149°F

(1) Switch to containment hot leg recirculation is assumed to be at six (6) hours from accident initiation.

Table D-8
MAAP-DBA ECCS and Recirculation Spray Piping Temperatures for Unit 2.

Case	Prior to Recirculation Spray System Start	Prior to Cold Leg Recirculation Transition	After Cold Leg Recirculation Transition	After Hot Leg Recirculation Transition(1)	RS HX Outlet Temp. After Cold Leg Recirculation(2)
1L	211°F	209°F	206.0°F	129°F	------
2L	202°F	199°F	196.4°F	129°F	------
2L1	204°F	202°F	200.6°F	134°F	------
2L2	201°F	198°F	195.9°F	130°F	------
3L	210°F	210°F	209.4°F	114°F	115°F

(1) Switch to containment hot leg recirculation is assumed to be at six (6) hours from accident initiation.

(2) Temperature of water supplied to safety injection piping in cold and hot leg recirculation modes assuming RSS train is secured if SW train failure occurs

Table D-9

MAAP-DBA Minimum Recirculation Spray and LHSI Pump NPSH Values for LOCA for Unit 1.

Description	Power Level, %	Single Failure	IRS NPSH (ft)	ORS NPSH (ft)	LHSI NPSH (ft)
1L-DEHL MIN SI	100.6	DG	22.0	26.7	---
1L1-DEHL MIN SI	100.6	LHSI	22.5	22.6	---
2L-DEHL MAX SI	100.6	None	22.5	22.8	---
3L-DEHL MIN SI	100.6	DG	26.2	25.2	---
4L-DEHL MAX SI	100.6	None	22.1	22.8	---
4L1-DEHL MAX SI	100.6	QS	16.8	21.0	---
6L-DEPS MIN SI	100.6	DG	15.4	14.7	16.0
7L-DEPS MAX SI	100.6	CIB	17.2	16.3	19.5
RS_12in_DG_HL(1)	100.6	DG	22.3	20.0	23.7
LHSI_05in_DG_HL(2)	100.6	DG	29.5	28.1	23.4

Single Failures – Failed Equipment

CIB	One train each, QSS, RSS
DG	One train each, SI, QSS, RSS
LHSI	One LHSI train
QS	One train of QS
(1)	Limiting case for RS pump from small and intermediate breaks
(2)	Limiting case for LHSI pump from small and intermediate breaks

Table D-10

MAAP-DBA Minimum Recirculation Spray Pump NPSH Values for LOCA for Unit 2.

Description	Power Level, %	Single Failure	RS NPSH (ft)
1L-DEPS MIN SI	100.6	DG	20.2
2L-DEPS MAX SI	100.6	CIB	20.4
2L2-DEPS MAX SI	100.6	RELAY	20.5
3L-DEHL	100.6	None	20.5
MSI_01IN_QS_CL(1)	100.6	QS	20.0

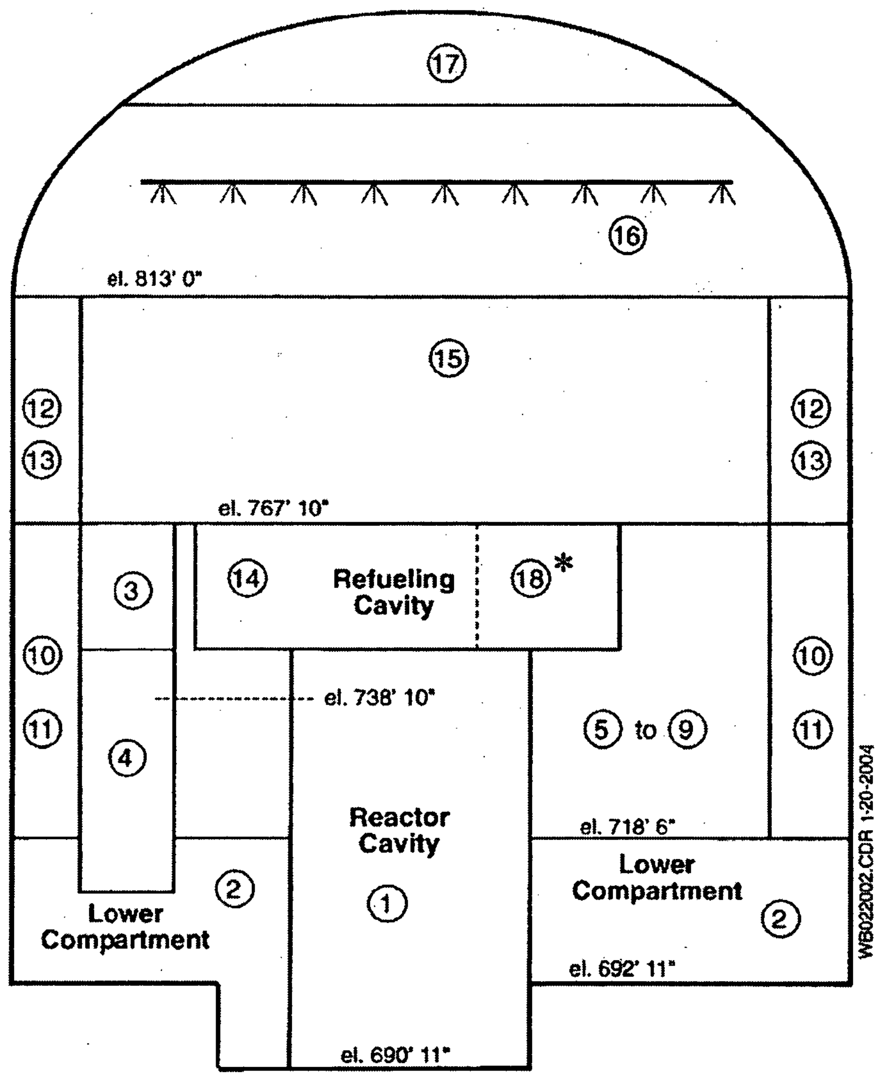
Single Failures – Failed Equipment

CIB One train each, QSS, RSS fail.

DG One train each, SI, QSS, RSS, and service water fail.

RELAY One train of RSS fails due to pump start relay failure.

(1) Limiting case from small and intermediate break sizes



*Refueling cavity is modeled as two nodes (14 and 18) in Unit 1 and a single node (14) in Unit 2.

Figure D-1:
MAAP-DBA Containment Nodalization for Units 1 and 2

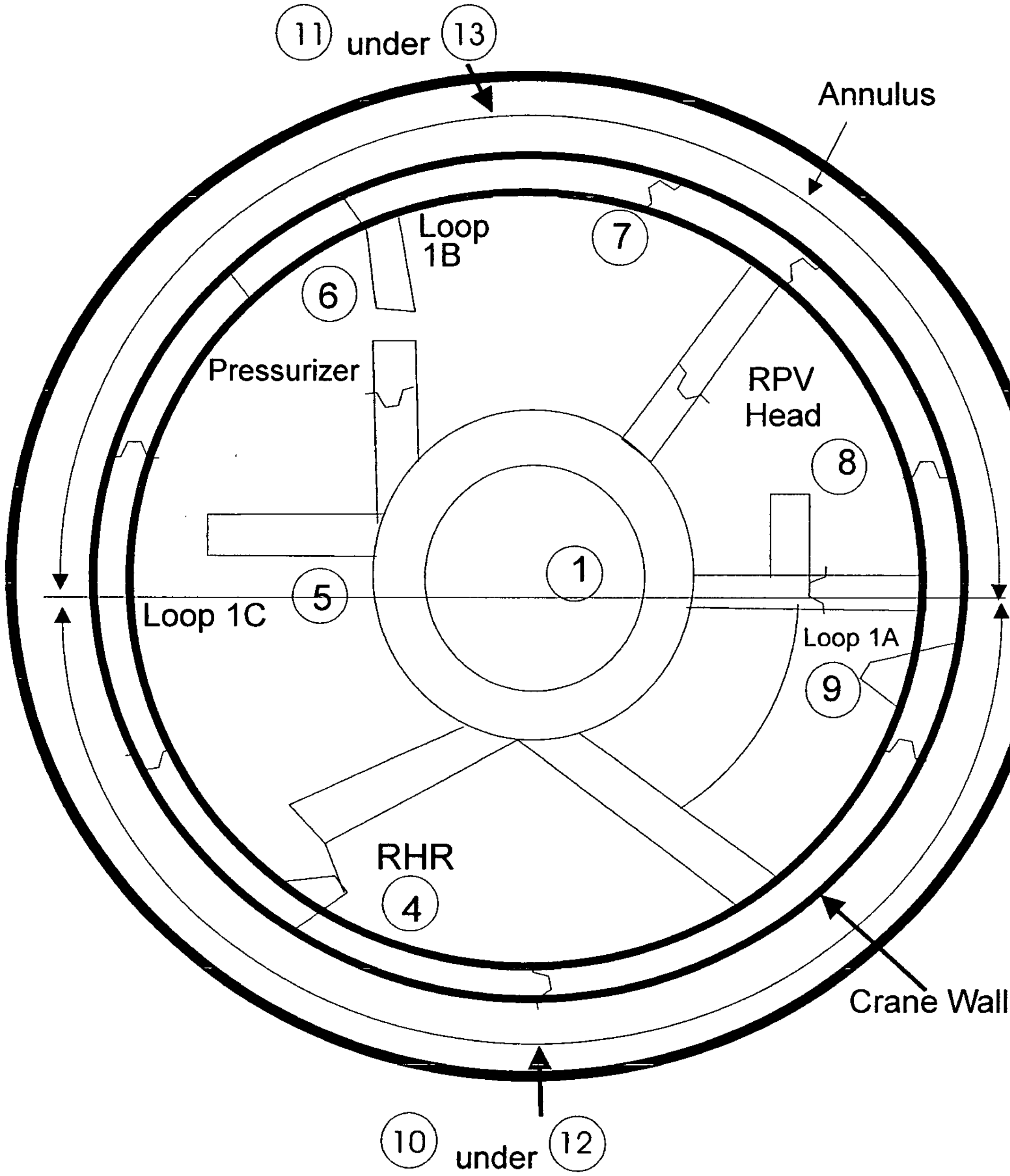
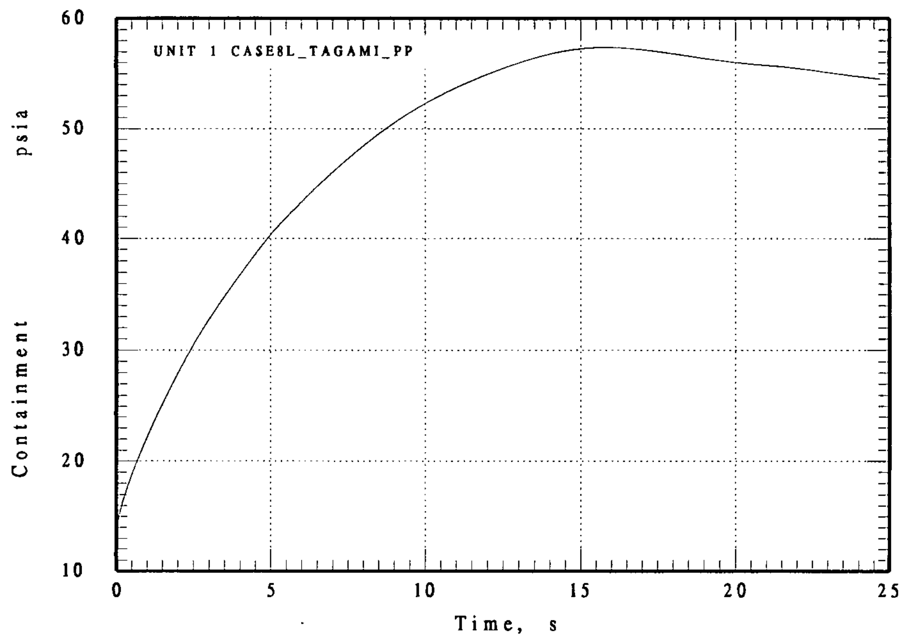


Figure D-2:
MAAP-DBA Containment Nodalization (Plan View) for Units 1 and 2



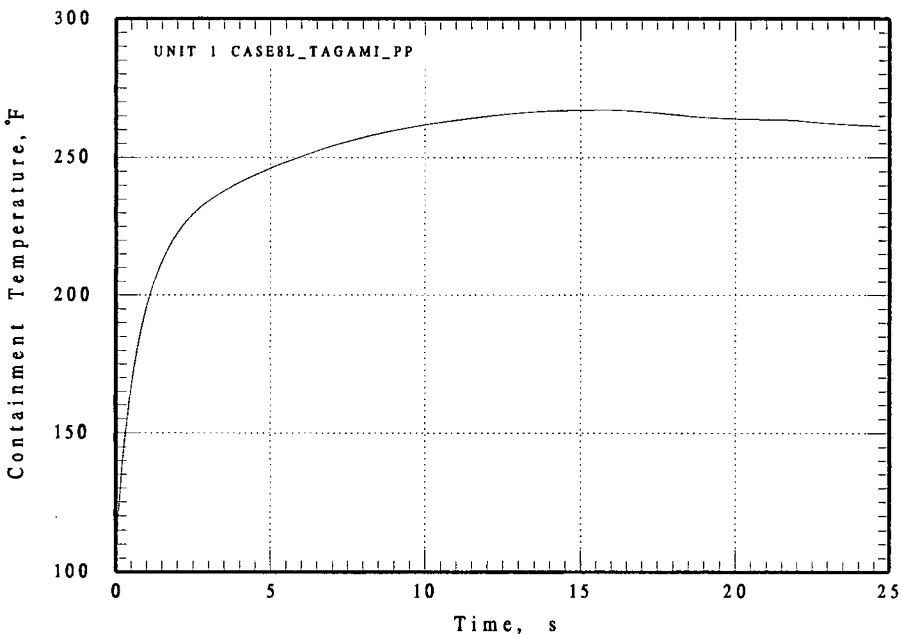
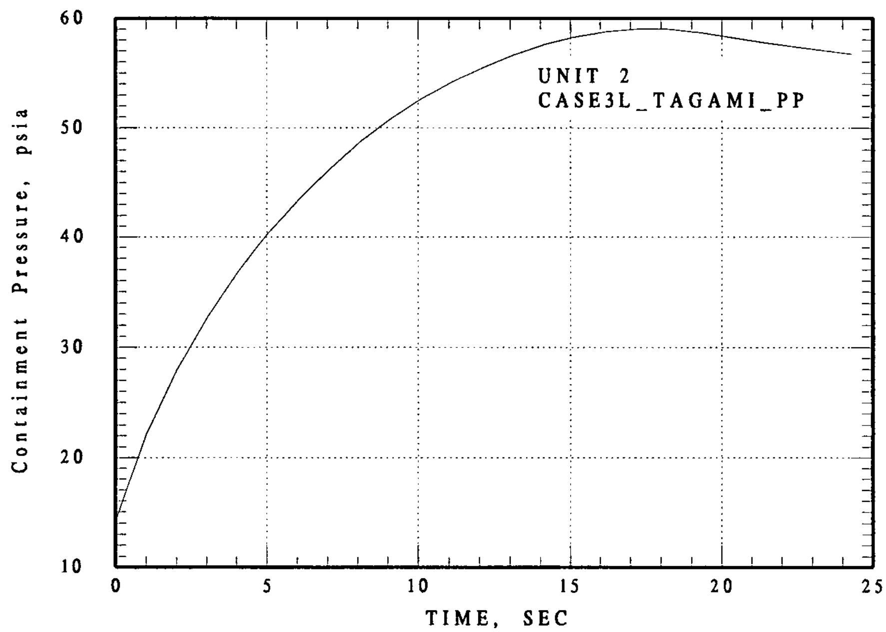


Figure D-3:

Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL (Case 8L) Analyses for Unit 1.



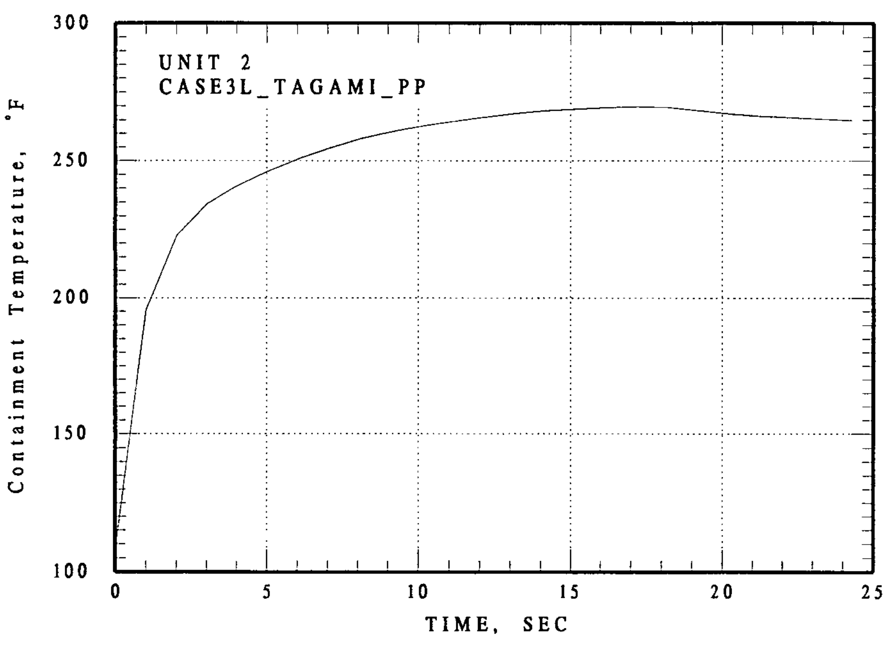


Figure D-4:

Containment Pressurization and Gas Atmosphere Temperature Histories for the DEHL (Case 3L) Analyses for Unit 2.

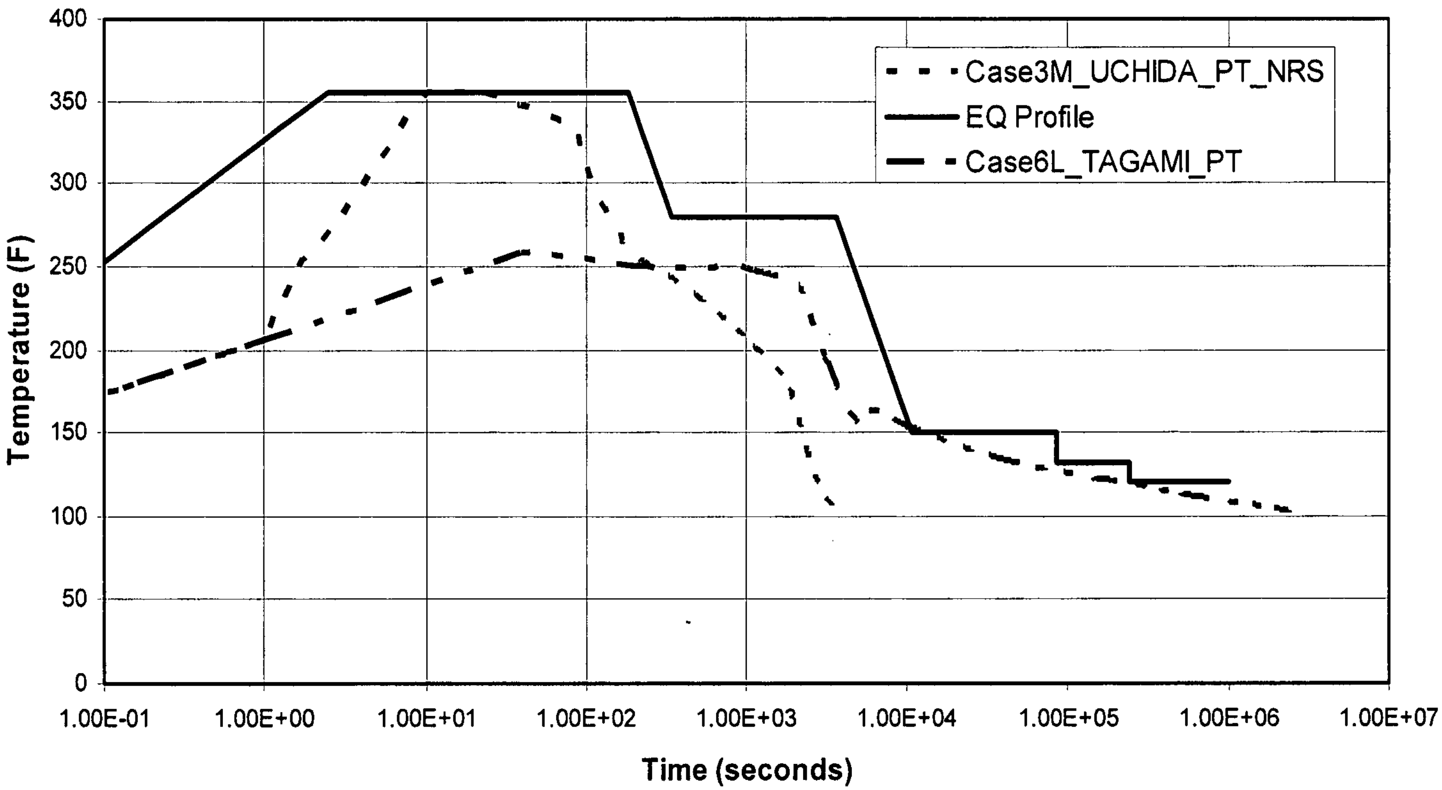


Figure D-5:
Containment Temperature Profiles LOCA and MSLB for Unit 1

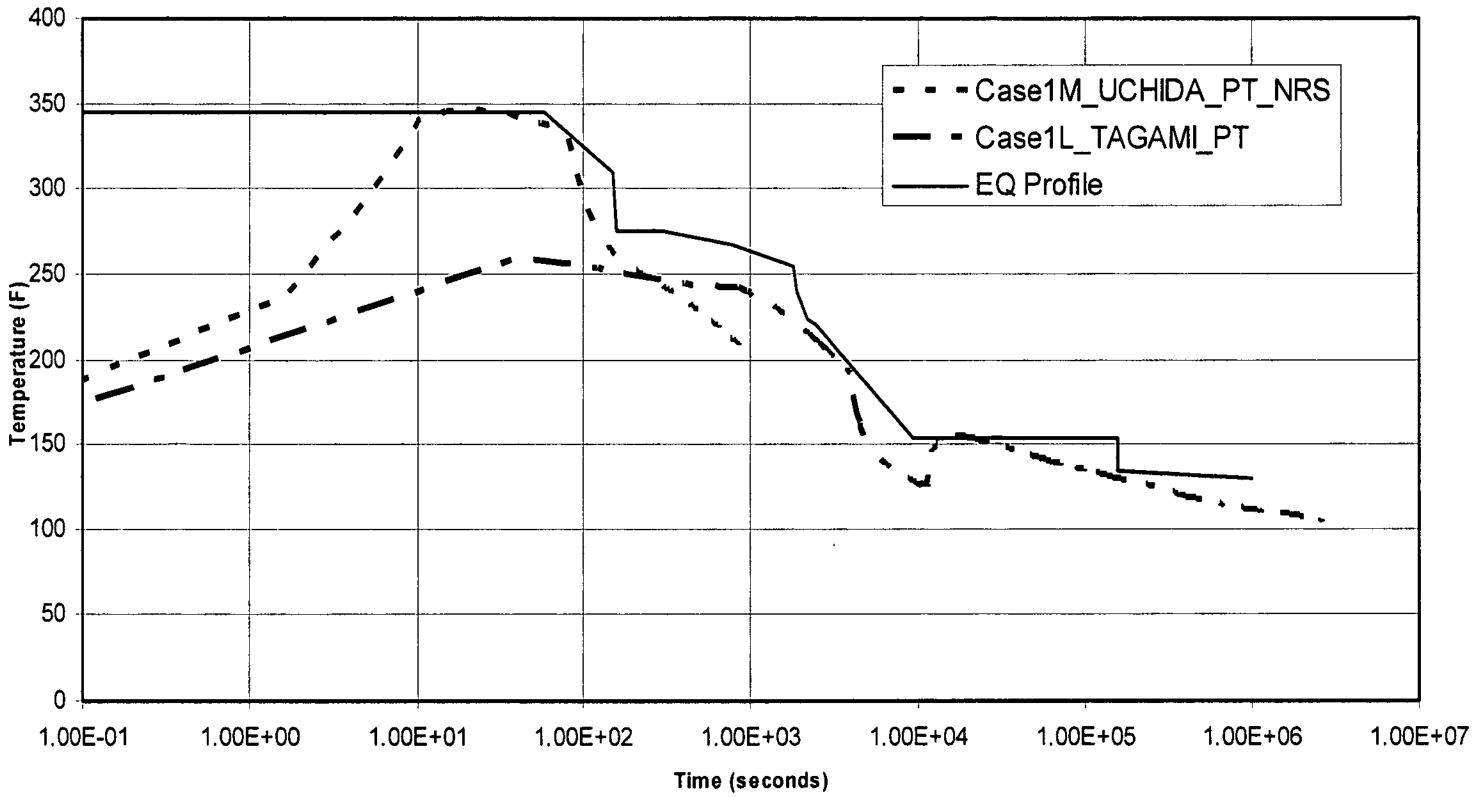


Figure D-6:
Containment Temperature Profiles LOCA and MSLB for Unit 2

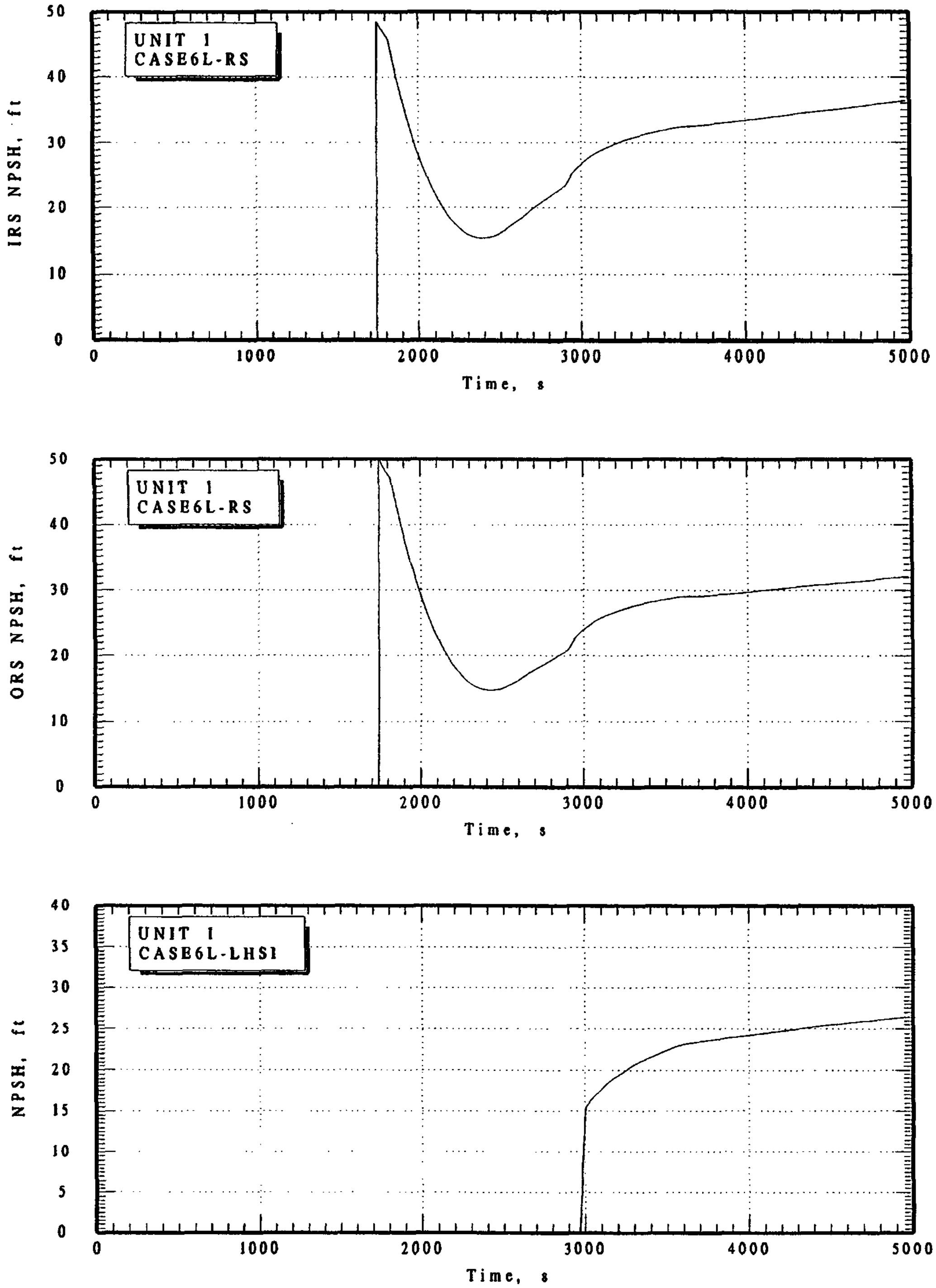


Figure D-7:
Available NPSH for Unit 1 RS and LHSI Pumps

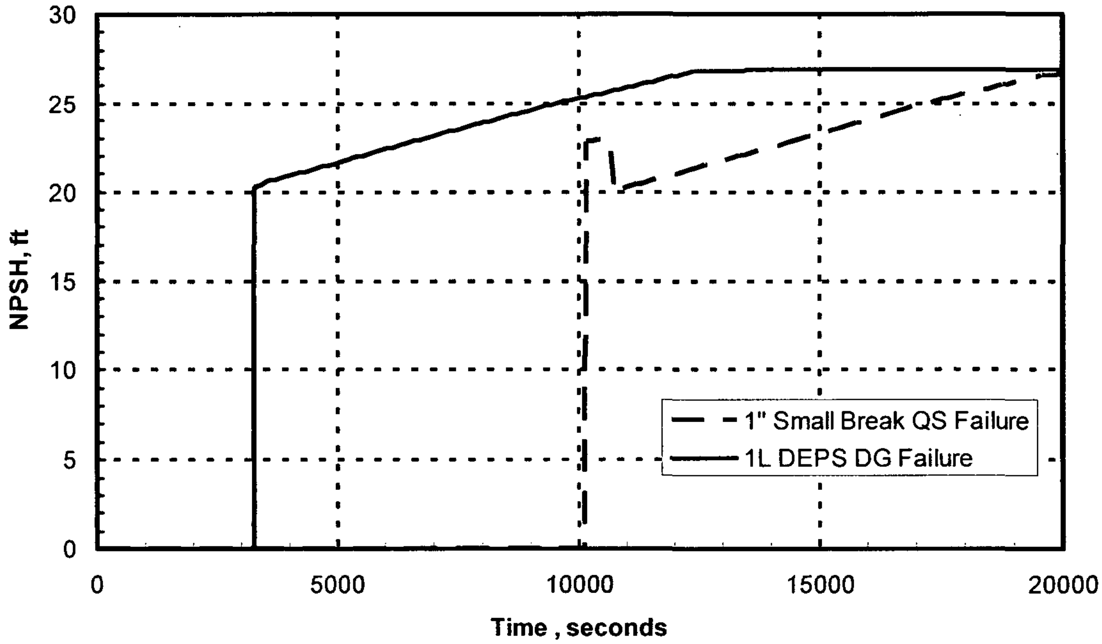


Figure D-8:
Available NPSH for Unit 2 RS Pumps

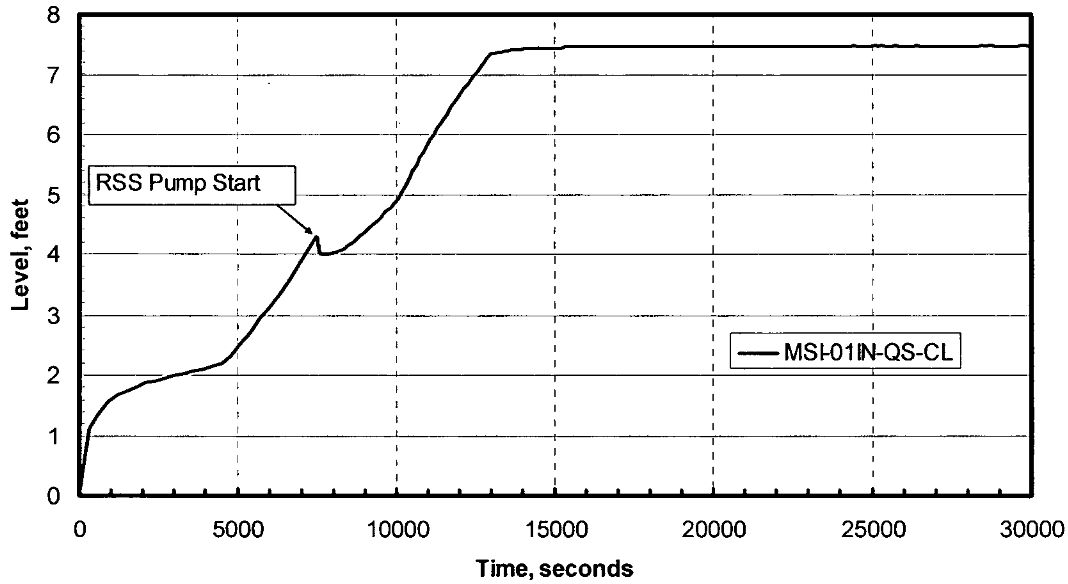


Figure D-9:
Limiting Sump Inventory at RS Pump Start for Unit 1

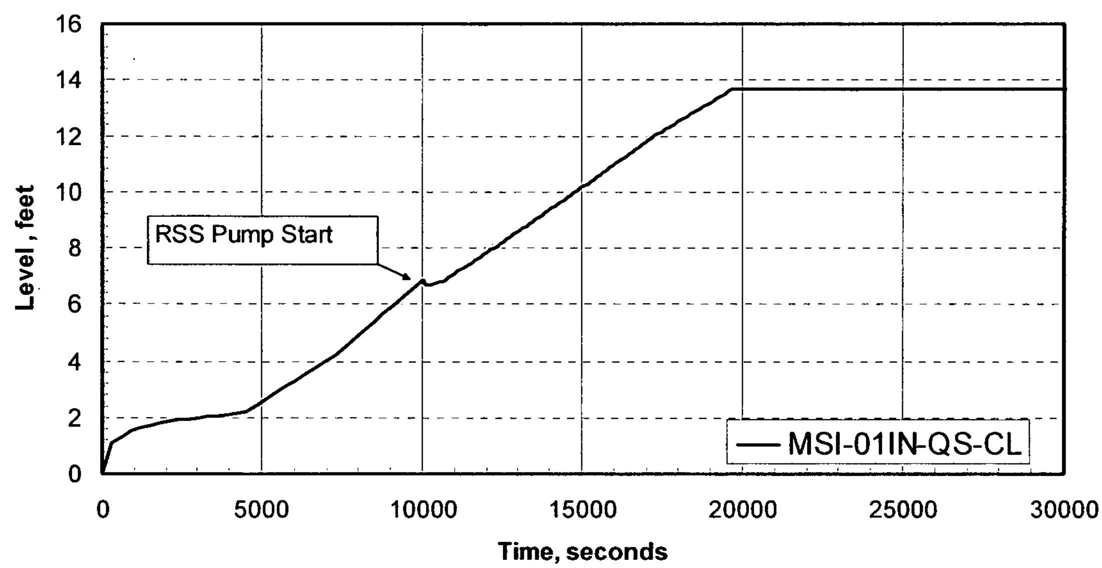


Figure D-10:
Limiting Sump Inventory at RS Pump Start for Unit 2

Enclosure Attachment E

Beaver Valley Power Station, Unit Nos. 1 and 2

License Amendment Requests 334 (Unit 1) and 205 (Unit 2)

Revised Dose Consequence Analyses

Loss-of-Coolant Accident

RADIOLOGICAL ASSESSMENT

Introduction

The current licensing basis methodology for estimating the dose consequences following a Loss of Coolant Accident (LOCA) at Beaver Valley Power Station (BVPS) is based on the utilization of Alternative Source Term (AST) methodology in accordance with 10CFR 50.67, SRP 15.0.1, and Regulatory Guide 1.183 (References 1, 2 and 3).

The licensing application summarizing the use of AST to estimate the dose consequences following a LOCA at BVPS was submitted to the NRC via License Amendment Request (LAR) Nos. 300 and 172 (Reference 6), and was approved by the NRC SER for Operating License Amendment Nos. 257 and 139 (Reference 11).

This attachment addresses the potential impact on the estimated dose consequences following a LOCA due to changes in the recirculation spray system operation incorporated as part of the resolution to GSI-191 and associated containment sump strainer modification. For purposes of the analysis documented in this attachment, and to be consistent with plant terminology, a Containment Pressure High-High signal is equivalent to a containment isolation phase B (CIB) signal.

The changes in design input associated with the above modification that impact the LOCA dose consequence analysis include:

- Delayed initiation of recirculation sprays
- Changes in the sump water volume vs time
- Change in sump water temperature at initiation of recirculation
- Changes in elemental iodine and aerosol fission product removal rates as a function of time
- Changes in the fractional volumetric release from the Refueling Water Storage Tank (RWST) gas space as a function of time

Regulatory Approach

Dose Acceptance Criteria

In accordance with current licensing basis, the acceptance criteria for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses for the LOCA 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 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.

In accordance with current licensing basis, the habitability of the BVPS Emergency Response Facility (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. 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 Emergency Response Facility (ERF).

Changes to Current Licensing Basis

The current licensing basis LOCA dose consequence analysis utilizes aerosol removal coefficients due to sprays and diffusiophoresis, based on containment conditions (i.e., containment pressure, temperature, relative humidity and steam condensation rates) that are reflective of a NUREG 1465 (Reference 16) delayed ECCS scenario, and are developed by S&W Proprietary code LOCTIC.

The updated LOCA dose consequence analysis reported herein utilizes aerosol removal coefficients due to sprays and diffusiophoresis, based on containment conditions (i.e., containment pressure, temperature, relative humidity and steam condensation rates) that are reflective of a traditional design basis LOCA, and are developed by EPRI code MAAP-DBA.

This change in design basis is conservative as it maximizes pressure and temperature and minimizes condensation rates during the source injection phase. It is primarily intended to maintain consistency in the thermal-hydraulic computer codes utilized by BVPS to support containment analyses vs dose consequence analyses.

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 to support nuclear power plant design and are a part of BVPS current licensing basis:

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

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

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

In addition, the following computer code has been used by Fauske and Associates to estimate the containment pressure, temperature, relative humidity and steam condensation rate which are used as input in the aerosol removal calculations.

4. EPRI Computer Code MAAP-DBA, Version 0, September 2003.

Radiation Source Terms

Core Activity Inventory

The current licensing basis equilibrium core inventory utilized to support this application is based on a core power level of 2918 MWt, and current licensed values of fuel enrichment and burnup. The methodology used to develop the core inventory, and the associated isotopic listing, is presented in Section 5.3.3.1 and Table 5.3.3-1 of LAR No's 300 and 172 (Reference 6), and was approved by the NRC SER for Operating License Amendment Nos. 257 and 139 (Reference 11).

This modification does not impact the licensing basis core inventory used for the dose consequence analyses. The design basis core activity of isotopes significant to dose consequences previously provided in Table 5.3.3-1 of Reference 6 is reproduced herein in Table E-1.

Technical Specification Primary Coolant Activity Inventory

The current licensing basis Technical Specification primary coolant concentrations utilized to support this application reflect an equilibrium core inventory based on a core power level of 2918 MWt, and current licensed values of fuel enrichment and burnup. The methodology to develop the Technical Specification primary coolant concentrations was discussed in Section 5.3.3.2 of LAR No's 300 and 172 (Reference 6), updated by Section 5.11.4.2 and Table 5.11.4-1 of LAR No's 302 and 173 (Reference 9), and approved by the NRC SER for Operating License Amendment Nos. 275 and 156 (Reference 14).

This modification does not impact the licensing basis noble gas and halogen primary coolant Technical Specification activity concentrations for BVPS previously provided in Table 5.11.4-1 of Reference 9 and are reproduced herein in Table E-2.

Accident Atmospheric Dispersion Factors

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 and are consistent with current licensing basis. These values are presented in Table E-3. For the purposes of performing bounding analyses representative for both units, the BVPS-2 EAB χ/Q's are utilized in assessing the impact of this modification on the post-LOCA dose consequences. As noted in Table E-3, the LPZ χ/Q's are the same for both Units.

On-Site Atmospheric Dispersion Factors

The Control Room and ERF atmospheric dispersion factors (χ/Q) for BVPS Unit 1 (BVPS-1) and BVPS Unit 2 (BVPS-2) remain unchanged by this application and are consistent with the current licensing basis. The (χ/Q) values for all Control Room and Emergency Response Facility release-receptor combinations applicable to the BVPS LOCA are summarized in Tables E-4 (BVPS-1) and E-5 (BVPS-2).

Dose Calculation Methodology

The dose calculation methodology is similar to that outlined in Section 5.3.5 of Reference 6 and is current licensing basis. As noted in Reference 6, 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 4) 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:

<u>Committed Effective Dose Equivalent (CEDE) Inhalation Dose</u> – The dose conversion factors by isotope 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 control room (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/m3)
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 fluence-to-dose factors" (Reference 5).

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 8) where 1 rad is assumed to be equivalent to 1 rem.

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

$E\gamma_{BAR}$	=	average gamma released per disintegration (Mev/dis)
$\psi(x,y,0)$	=	concentration time integral (Ci-sec/m3)
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
3.7x1010 (dis/sec-Ci)	=	disintegration rate per curie
1293 (g/m3)	=	density of air at S.T.P.
100	=	ergs per gram per rad
2	=	factor for converting an infinite to a semi-infinite cloud

Radiological Accident Analyses

The changes in the recirculation spray system operation proposed by this application will potentially impact the reported dose consequences of the BVPS-1 and BVPS-2 LOCA. The radiological consequences of the remaining design basis accidents are not impacted by this application.

Loss of Coolant Accident (LOCA)

Regulatory Guide 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.

For BVPS four activity release paths following a LOCA have been identified: (a) Containment Vacuum System Release, (b) Containment Leakage (c) Engineered Safety Features (ESF) System Leakage and d) RWST back leakage.

The Unit 1 LHSI pump mini-flow lines go back to the RWST, however, they are isolated following transfer to recirculation mode and therefore never pass contaminated water. The Unit 2 RSS mini-flow lines are routed back to the containment sump. The HHSI pump min-flow lines at both units connect back to the pump suction such that the flow never leaves the system. Leakage from the ECCS system to the RWST is accounted for in the analysis. The assumption of two times the allowed leakage (1 gpm) is used.

Table E-7 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.

Doses due to Submersion and Inhalation

Stone & Webster (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.

Containment Vacuum System Release

In accordance with current licensing basis, 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. Note that the dose consequences of this release path are not impacted by this modification. The methodology utilized to develop the dose consequences via this release path was discussed in LAR No's 300 and 172 (Reference 6) and is repeated herein.

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 (assumed to be at containment design pressure), 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.

Containment Leakage

The inventory of fission products in the reactor core available for release via containment leakage following a LOCA is based on Table E-1 which represents a conservative equilibrium reactor core inventory of dose significant isotopes, assuming maximum full power operation 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. The BVPS design includes a containment quench spray and a containment recirculation 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 RWST, into the containment, via the quench spray system spray headers. Based on an assumption of a Loss of Offsite Power (LOOP) coincident with the LOCA, the quench spray is assumed to be initiated, at either unit, by approximately 85 seconds, and is available until depletion of the RWST inventory.

As a result of this modification, recirculation spray starts on RWST level; consequently the recirculation spray initiation time is dependent on the accident scenario. For purposes of conservatism, the model is based on a scenario that maximizes the delay in start of the recirculation spray. Recirculation spray is assumed to start at 3870 secs (value based on BVPS-2, the BVPS-1 value is 2078 secs). The recirculation spray system 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 96 hrs 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

In accordance with current licensing basis, the particulate removal rate from the containment atmosphere by the quench and recirculation spray systems is calculated with Stone & Webster's proprietary SWNAUA Computer Program (Reference 10). The SWNAUA code is discussed in LAR No's 300 and 172 (Reference 6). As noted in Reference 6, the SWNAUA Program is a derivative of the NAUA/MOD4 Computer Program (Reference 13). 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 (DBA) calculations. As discussed in Reference 6, the model correlations that were implemented into SWNAUA tend to underestimate the spray removal coefficient. 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.

As noted in Reference 6, 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 plateout, 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, gravitational settling of aerosols is credited in the unsprayed region.

The spray model utilized for the LOCA analysis including a description of the aerosol and the aerosol injection rate is provided in Section 5.3.6.3.1.2 of LAR No's 300 and 172 (Reference 6) and was approved by NRC in NRC SER for Operating License Amendment Nos. 257 and 139 (Reference 11).

The plant parameters that are utilized herein to develop the aerosol removal coefficients are bounding for both Beaver Valley Units and 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	1821 gpm (120 - 2080 sec) 2,956 gpm (2080-3,870 sec) 6,108 gpm (3,870-10,938 sec) 3,267 gpm (10,938 – 346,000 sec)
Spray droplet radius	500 microns

The containment pressure, temperature, steam condensing rate and relative humidity transients utilized for the development of the aerosol removal coefficients are obtained from the MAAP model of the traditional design basis LOCA, and are presented in Table E-8. Since higher containment pressure and temperature, lower steam condensing rates and lower spray flow rates result in lower aerosol removal rates, the bounding outputs of two design basis LOCA cases per unit are conservatively utilized; i.e., a) Pump Suction Double Ended Rupture (PSDER) with failure of one diesel generator, and b) PSDER with CIB failure.

Removal of Particulates by Diffusiophoresis

As noted in Reference 6, 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 bounding pressure, temperature and steam condensation rates used to develop the particulate removal coefficients due to diffusiophoresis are presented in Table E-8.

The coefficient for removal of particulates from the effectively sprayed and unsprayed regions of the containment are plotted versus time in Figures E-1 and E-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 by Sprays and Plate-out

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 7). 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 7). 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

In accordance with current licensing basis, 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.

For purposes of spray coverage, the quench spray is credited prior to initiation of recirculation spray. Subsequent to initiation of recirculation spray, no credit is taken for spray coverage by quench spray operation.

The effective spray coverage fraction is conservatively assessed to be 63.0 percent of the containment free volume at either unit for the duration of time that credit is taken for sprays. 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 sprays.

Containment Mixing

Mixing in the containment following a postulated design basis LOCA 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.

In accordance with current licensing basis, 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, and in accordance with current licensing basis, 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: (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 and current licensing basis, 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, in accordance with current licensing basis, 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 Group:	Te, Sb, Se, Sn, In, Ge, Ga, Cd, As, Ag
Ba,Sr:	Ba, Sr, Ra
Noble Metals:	Ru, Rh, Pd, Mo, Tc, Co
Cerium Group:	Ce, Pu, Np, Th, U, Pa, Cf, Ac
Lanthanides:	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am, Gd, Ho, Tb, Dy

In accordance with Regulatory Guide 1.183, since the BVPS long term sump pH is controlled to values of 7 and greater, the chemical form of the radio iodine 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 flow rates 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.

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. For BVPS, this DF value is reached for elemental iodine at approximately 6.4 hrs after the accident.

Mixing between the "effectively" sprayed and unsprayed regions of the containment is assumed for the duration of the accident. 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 (HCl 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 19).

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.

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 321,155 gallons (limiting parameter values based on BVPS-1) 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 40% of the final value. For the next one and a half hours the sump volume is about 56 % 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 halogens, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. The subsequent environmental radioactivity release is discussed below:

- <u>ESF leakage</u>: Equipment carrying sump fluids and located outside containment are postulated to leak at twice the surveillance limit 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 is conservatively assumed to start at 1200 seconds (bounding for both units). 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 peak sump water temperature after 20 minutes is 250°F. 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.

- <u>RWST Back-leakage</u>: Sump water back-leakage into the RWST (located in the Yard) is postulated to occur at twice the surveillance limit of 1 gpm and is released to the environment via the RWST vent. 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., time dependent iodine partition coefficient)

  For the limiting BVPS-1 case, sump water begins to leak into the RWST at 1782 seconds after the LOCA. At 3055 secs (changed from the previous 5178 secs), the iodine begins to flow out of the RWST and disperses to the environment. Environmental airborne iodine activity resulting from RWST leakage is assumed to be 97% elemental and 3% organic.

In accordance with current licensing basis, in the dose model, this phenomenon is modeled using a series of effective environmental release rate lambdas from the RWST vent.

For purposes of limiting the amount of re-analyses, and as a conservative approximation, the current design basis RWST iodine release fractions provided in Figure E-3 (previously approved by NRC via Reference 11) are multiplied by 2 and used herein with the following adjustment. The release fraction (times two) applicable at T=5178 secs (see Figure E-3) is also applied to the time period between T=3055 secs and T=5178 secs. This approach has been determined to be very conservative based on confirmatory analyses performed to estimate the RWST iodine release fractions based on the updated transient.

Site Boundary Dose Assessment

The LOCA dose consequence analysis reflects bounding parameter values to encompass an event at either unit. The worst 2-hour period dose at the EAB, and the dose at the LPZ for the duration of the release, is calculated 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.

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 are presented in Table E-9.

Control Room Design/Operation/Transport Model

The BVPS control room design and operation outlined in LAR No's 302 and 173 (Reference 9) remains unchanged by this application and is summarized below.

Beaver Valley Power Station 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 providing a total supply of 500 cfm of unfiltered outside air makeup which includes all potential in-leakage and uncertainties.

Since the BVPS control rooms are contained in a single control room envelope, they are 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, the most limiting χ/Q associated with the release point/receptor for an accident in either unit is utilized.

The control room post-accident ventilation model utilized in the LOCA dose analysis corresponds to an assumed "single intake" which utilizes the worst case atmospheric dispersion factor (χ/Q) from release points associated with an accident 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, at BVPS-1 and BVPS-2, are provided in Tables E-4 and E-5.

In accordance with the current licensing basis, the atmospheric dispersion factors associated with control room inleakage is assumed to be the same as that utilized for the control room intake. The 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 inleakage is potentially 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 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.

Current plant design will automatically isolate the control room and initiate control room pressurization via the BVPS-2 control room emergency ventilation system (CREVS) upon receipt of a Containment Pressure High-High signal (equivalent to CIB) from either unit, or a high radiation alarm from the control room area monitors. The BVPS-2 CREVS system is safety-related, fully automated, and fully compliant with all relevant regulatory requirements. 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 BVPS2 CREVS train for maintenance.

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 17), Generic Letter 99-02 (Reference 18), ASTM D3803-89 and ANSI N510-1980.

Taking into account LOOP, the maximum estimated delay in attaining 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 diesel start, sequencing and damper movement/re-alignment. The CREVS Train A fan is expected to get a start signal at T=90 secs. Considering the CREVS time delay relay setting for Train B fan start, plus fan acceleration time, the total auto start delay is estimated to be 137 sec.

Since the analysis is bounding for an event at either unit, no credit is taken for automatic initiation of the BVPS-2 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 minutes.

The dose model conservatively assumes 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 15) as well as a margin to allow for potential future deterioration.

The control room emergency filtered ventilation intake flow varies between 600 to 1030 cfm, which includes allowance for measurement uncertainties. 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, and to also accommodate margin for potential future deterioration.

In accordance with current licensing basis, and 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 E-6 lists key assumptions/parameters associated with BVPS control room design utilized for the LOCA dose analyses.

In accordance with current licensing basis, 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.

In addition, and in accordance with current licensing basis, 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 20). 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 0 to 30-day dose to an operator in the control room due to airborne radioactivity releases is developed and 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.

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 foot 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 estimated 30 day dose to the operator in the control room following a LOCA at either unit is reported in Table E-10.

ERF Habitability

In accordance with current licensing basis and 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.

Doses due to Submersion and Inhalation

As discussed in Reference 6, Beaver Valley Power Station is served by a single 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, and in accordance with the current licensing basis model, 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 E-11 lists key parameters associated with the ERF design.

<u>ERF Dose due to Direct Shine from External Sources</u>

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.

<u>30 day Dose in the ERF.</u>

The maximum estimated 30 day dose to the operator in the ERF following a LOCA at either unit is reported in Table E-10. Note that the conclusion noted in Reference 6 that neither the SLCRS filters, nor the ERF filters are required to maintain ERF Habitability continues to be valid.

Summary of Results

The radiological analyses and evaluations developed in support of this application demonstrate that the changes in recirculation spray system operation incorporated as part of the resolution to GSI-191 and associated containment sump strainer modification, will not impact compliance with applicable regulatory requirements.

The LOCA has been evaluated using the guidance provided in Regulatory Guide 1.183. As demonstrated in Tables E-9 ad E-10 the estimated dose consequences at the EAB, LPZ, Control Room remain within the acceptance criteria of 10CFR50.67 as supplemented by Regulatory Guide 1.183 and SRP 15.0.1. In addition, as noted in Table E-10, the analyses and evaluations also demonstrates that the dose consequences in the Emergency Response Facility remain compliant 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.

References

1. 10CFR50.67, “Accident Source Term.”

2. NUREG-0800, Standard Review Plan 15.0.1, “Radiological Consequence Analyses using Alternative Source Terms,” Revision 0.

3. Regulatory Guide 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

4. EPA-520/1-88-020, September 1988, Federal Guidance Report No.11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion.”

5. ANSI/ANS 6.1.1-1991, “Neutron and Gamma-ray Fluence-to-dose Factors.”

6. Beaver Valley Power Station Units 1 and 2 Licensing Amendment Request (LAR) No’s 300 and 172, L-02-069 entitled “Containment Conversion” June 5, 2002.

7. NUREG-0800, 1988, Standard Review Plan, “Containment Spray as a Fission Product Cleanup System,” Section 6.5.2, Revision 2.

8. TID-24190, Air Resources Laboratories, "Meteorology and Atomic Energy," July 1968.

9. Beaver Valley Power Station Units 1 and 2 Licensing Amendment Request (LAR) No's 302 and 173, L-04-125, Subject: Extended Power Uprate, October 4, 2004.

10. Lischer, D.J., User Manual, Aerosol Behavior in a Condensing Atmosphere (SWNAUA), June 1993, (Proprietary).

11. NRC Safety Evaluation Report Enclosing Amendment No. 257 (BVPS-1) and No. 139 (BVPS-2), "Beaver Valley Power Station Units 1 & 2 – Issuance of Amendments Re: Selective Implementation of Alternate Source Terms and Control Room Habitability Technical Specification Changes," September 10, 2003.

12. NUREG-0737, Supplement 1, "Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability," December 17, 1982.

13. Bunz, H., Kayro, M., Schöck, W., 1982, NAUA/Mod4 - A Code for Calculating Aerosol Behaviour in LWR Core Melt Accidents, Code Description and User Manual, KfK.

14. NRC Safety Evaluation Report Enclosing Amendment No. 275 (BVPS-1) and No. 156 (BVPS-2), "Beaver Valley Power Station Units 1 & 2 – Issuance of Amendments Re: Extended Power Uprate," July 19, 2006.

15. NUREG-0800, SRP 6.4, Revision 2, "Control Room Habitability System."

16. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants", February 1995.

17. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 2.

18. USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal," NRC Generic Letter 99-02, June 3, 1999.

19. NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents – Final Report," April 1992.

20. NRC Information Notice 93-17, Revision 1, "Safety Systems Response to Loss of Coolant and Loss of Offsite Power," March 25, 1994 (original issue March 8, 1993).

Table E-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 E-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 E-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 E-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 E-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 E-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 E-2 BVPS Primary Coolant Technical Specification Iodine and Noble Gas Concentrations	
Nuclide	**Primary Coolant (μCi/gm)**
I-131	2.74E-01
I-132	1.08E-01
I-133	4.10E-01
I-134	6.00E-02
I-135	2.36E-01
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 E-3 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 E-4
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 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 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 E-5
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 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 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 E-6 Analysis Assumptions & Key Parameter Values BVPS Common Control Room		
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 E-7 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	85.4 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 E-1 & E-2
Core Activity	Table E-1
Release Point	Worst cases between Containment Outer Wall & SLCRS Vent (Containment Top)

Table E-7 (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)	19,111 ft^3 (1.13E6 lbm) 25,333 ft^3 (1.51E6 lbm) 43,577 ft^3 (2.68E6 lbm)
ESF Leakage		
ESF Leakrate		11400 cc/hr (2×expected)
Leakage Period		1200 sec 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
RWST (backleakage)		
Sump water back-flow to RWST		2 gpm (includes factor of 2 margin)
Onset of back-leakage		1782 seconds
Onset of RWST activity venting		3055 seconds
Iodine Release Fraction via RWST		Figure E-3 values times 2
Release Point		RWST Vent
Containment Vacuum System Parameters		
Primary Coolant Tech Spec Activity		Table E-2
Chemical Form of Iodine Released		97% elemental; 3% organic
Containment Vacuum System Release		2200 scfm (5 sec release)
Release Point		Worst case between Containment Outer Wall and 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 E-8
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
0.0	14.2	0.0	108.0	0.0	0.0	0.0	0.500
17.3	56.5	17.3	266.7	40.0	277315.0	5.0	0.579
40.1	52.3	40.1	258.9	80.1	120655.9	10.0	1.000
80.2	51.5	80.2	256.0	120.2	101211.8	345600.0	1.000
120.6	50.0	120.6	253.2	160.2	84363.2		
160.2	49.0	160.2	251.3	200.2	74004.4		
200.2	48.3	200.2	250.0	240.2	66345.6		
240.2	47.8	240.2	249.4	280.7	61909.1		
280.7	47.6	280.7	248.9	321.0	58435.7		
321.0	47.4	321.0	248.5	361.0	55660.3		
361.0	47.4	361.0	248.4	401.0	53361.2		
401.0	47.3	401.0	248.3	441.0	51477.6		
441.0	47.3	441.0	248.3	481.0	49833.9		
481.0	47.4	481.0	248.3	522.1	48459.7		
522.1	47.5	522.1	248.4	562.1	47186.3		
562.1	47.6	562.1	248.6	602.1	47106.8		
602.1	47.7	602.1	248.9	642.1	46142.9		
642.1	47.9	642.1	249.2	682.1	45339.0		
682.1	48.1	682.1	249.5	722.1	44671.6		
722.1	48.2	722.1	249.8	762.1	44046.5		
762.1	48.4	762.1	250.1	802.1	43450.6		
802.1	48.6	802.1	250.5	842.1	40987.5		
842.1	48.9	842.1	250.9	882.1	39002.2		
882.1	48.6	882.1	250.4	922.1	36625.6		

Table E-8 (Continued)
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
922.1	48.3	922.1	249.8	962.1	35511.8		
962.1	48.0	962.1	249.2	1002.1	34577.7		
1002.1	47.7	1002.1	248.6	1042.1	33760.2		
1042.1	47.4	1042.1	248.1	1082.1	33031.9		
1082.1	47.2	1082.1	247.7	1122.1	32376.9		
1122.1	47.0	1122.1	247.3	1162.1	31885.7		
1162.1	46.8	1162.1	246.9	1202.1	31344.1		
1202.1	46.6	1202.1	246.6	1242.1	32338.9		
1242.1	46.5	1242.1	246.3	1282.1	31822.6		
1282.1	46.3	1282.1	246.0	1322.1	31350.0		
1322.1	46.2	1322.1	245.7	1362.1	30936.6		
1362.1	46.1	1362.1	245.4	1402.1	30555.2		
1402.1	45.9	1402.1	245.2	1442.1	30201.4		
1442.1	45.8	1442.1	244.9	1482.1	29871.6		
1482.1	45.7	1482.1	244.7	1522.1	29563.9		
1522.1	45.6	1522.1	244.5	1562.1	29183.4		
1562.1	45.4	1562.1	244.1	1602.1	28593.4		
1602.1	45.2	1602.1	243.6	1642.1	28094.8		
1642.1	45.0	1642.1	243.2	1682.1	27661.6		
1682.1	44.8	1682.1	242.8	1722.1	27262.5		
1722.1	44.6	1722.1	242.4	1800.1	26465.3		
1762.1	44.4	1762.1	242.0	2004.0	25498.4		
1804.0	44.2	1804.0	241.6	2766.0	22197.8		
1844.0	44.0	1844.0	241.2	3006.0	18874.7		

Table E-8 (Continued)
Containment Thermodynamic Data – Loss of Coolant Accident

Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
1884.0	43.8	1884.0	240.8	3221.1	18440.8		
1924.0	43.6	1924.0	240.5	3248.4	19906.7		
1964.0	43.5	1964.0	240.1	3548.4	10768.3		
2004.0	43.3	2004.0	239.8	4049.5	13086.0		
2044.0	43.1	2044.0	239.4	4359.5	13219.0		
2078.0	43.0	2078.0	239.1	4759.5	8477.6		
2093.0	43.0	2080.0	239.1	4959.5	10098.2		
2140.0	42.8	2081.0	239.3	5059.5	10485.0		
2170.0	42.1	2100.0	239.1	5159.5	10929.1		
2212.0	41.0	2152.4	238.2	5259.5	11132.0		
2252.0	40.1	2200.0	234.3	5363.8	5848.9		
2350.0	37.9	2252.0	231.3	5463.8	8500.4		
2450.0	35.9	2966.0	204.2	5563.8	9584.6		
2550.0	34.2	3609.5	197.5	5663.8	10343.7		
2650.0	32.6	3877.5	192.8	5763.8	10871.0		
2750.0	31.3	3955.5	190.0	5863.8	11164.7		
3446.5	29.0	4749.5	180.1	6563.8	11990.7		
3849.5	27.4	4849.5	183.7	6663.8	12075.3		
3955.5	27.0	5749.5	191.2	7163.8	12071.2		
4749.5	24.9	6749.5	191.4	7324.9	10980.8		
4849.5	25.3	7263.9	192.6	10025.8	10025.0		
5949.5	27.2	18074.7	188.9	10338.5	10042.1		
6749.5	27.2	35998.0	180.5	10938.5	10167.3		
7263.9	27.4	72198.0	170.9	11054.8	5388.5		

Table E-8 (Continued) Containment Thermodynamic Data – Loss of Coolant Accident							
Time (sec)	Pressure (psia)	Time (sec)	Temp (°F)	Time (sec)	Steam Condensing Rates (gm/sec)	Time (sec)	Relative Humidity
18074.7	26.8	144020.0	162.6	11154.8	6717.8		
35998.0	24.9	259020.0	155.3	11254.8	7739.4		
72198.0	23.1	346020.0	150.4	11354.8	8311.9		
144020.0	21.7			11454.8	8594.5		
259020.0	20.8			11554.8	8780.4		
346020.0	20.2			11569.0	5020.2		
				11682.9	6267.3		
				11782.9	7467.5		
				11882.9	8111.2		
				13882.9	9063.9		
				15082.9	8838.6		
				27200.0	7407.0		
				36000.0	6962.0		
				72000.0	5866.0		
				86400.0	5609.0		
				142920.0	5077.0		
				335920.0	4044.0		
				345920.0	3942.0		

Table E-9 Exclusion Area Boundary and Low Population Doses (TEDE)			
Accident	**EAB Dose (rem) [1,3]**	**LPZ Dose (rem) [2]**	**SB Reg. Limit (rem)**
LOCA	16.5	3.0	25

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 LOCA is 0.5 to 2.5 hrs

Table E-10 30 Day Integrated Control Room & ERF Doses (TEDE)		
	Control Room Operator	
Accident	**Dose (rem) [1]**	**Reg. Limit (rem)**
Control Room	2.5 (0.61)	5.00
Emergency Response Facility	3.5 (0.25)	5.00

Notes:

1. Portion shown in parenthesis represents that portion of the total dose that is the contribution of direct shine from contained sources/external cloud.

Table E-11 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

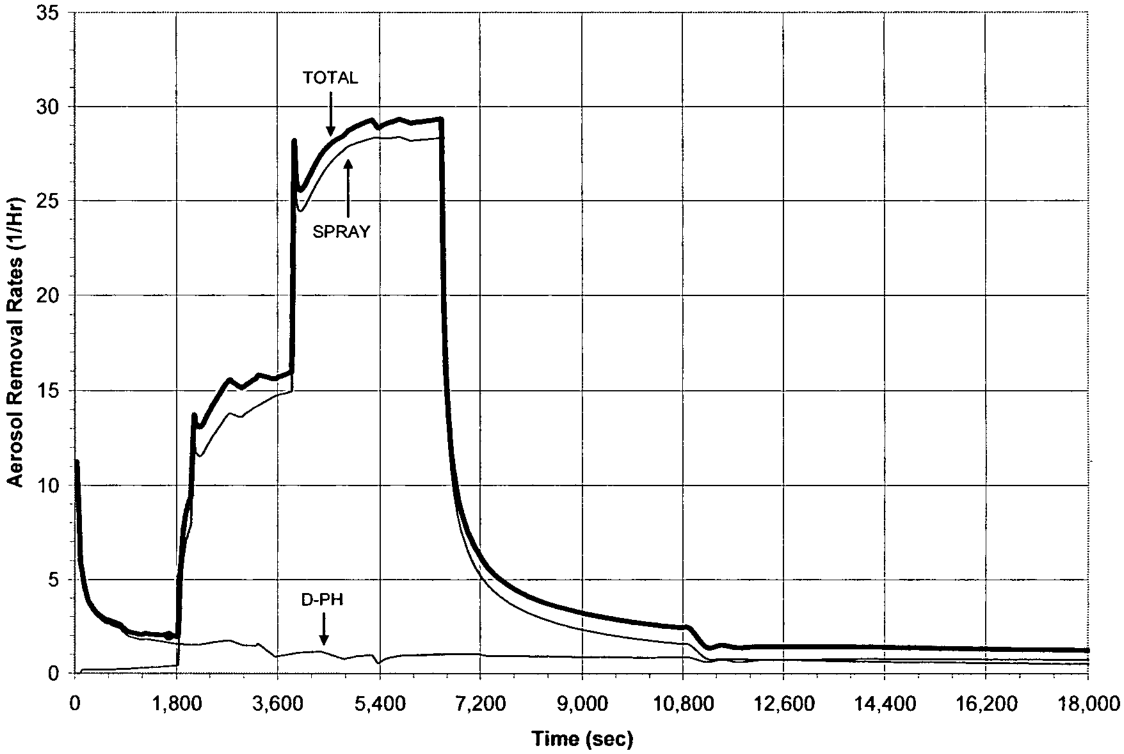


Figure E-1
Aerosol Removal Rates Within Sprayed Region (Design Basis LOCA CASE)

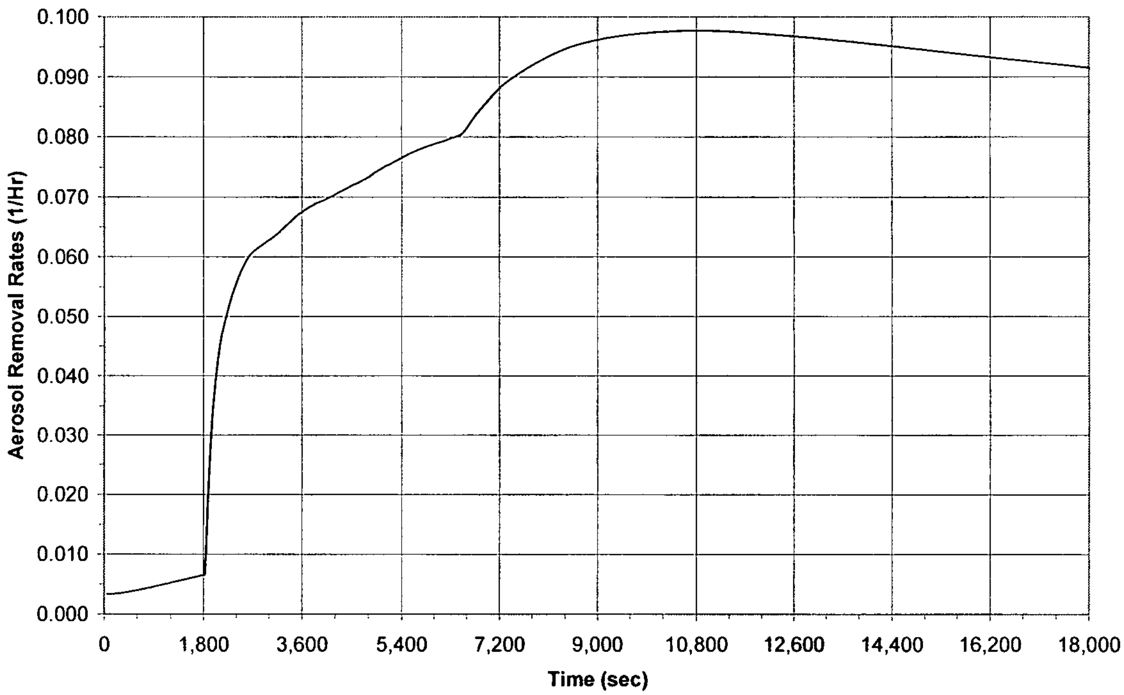
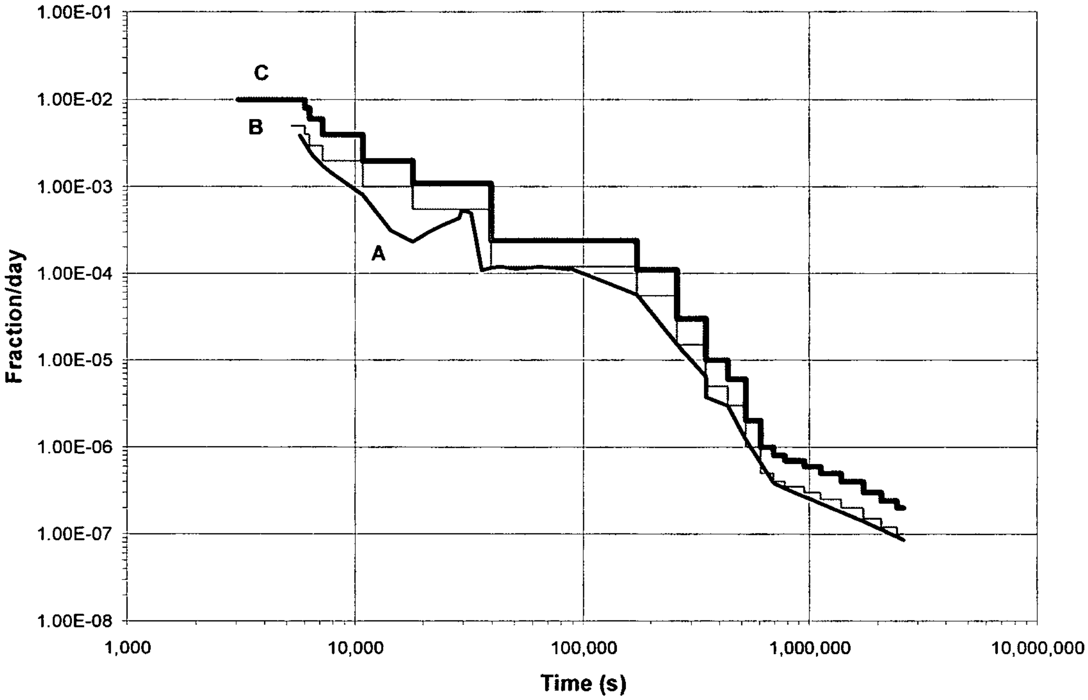


Figure E-2
Aerosol Removal Rates Within Unsprayed Region (Design Basis LOCA CASE)



Legend:

A: RWST Iodine Release Fractions calculated in support of Reference 11 (presented previously in Figure 5.3.6-3 of Reference 11)

B: RWST Iodine Release Fractions conservatively utilized in Reference 11 for purpose of estimating dose consequences

C: RWST Iodine Release Fractions conservatively utilized herein for purposes of estimating dose consequences

Figure E-3
RWST Iodine Release Fractions

Enclosure Attachment F

Beaver Valley Power Station, Unit Nos. 1 and 2

Commitment Summary

License Amendment Requests 334 (Unit 1) and 205 (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. If there are any questions, or if additional information is required, please contact Mr. Henry Hegrat, Supervisor- FENOC Fleet Licensing, at (330) 315-6944.

COMMITMENT	REFERENCE	DUE DATE
1. Compare the available NPSH margin for the recirculation spray system pumps to the head loss across the containment sump strainer.	Enclosure 1 Section 4.1.2	Following completion of each unit's strainer testing with the plant specific debris loading, but prior to implementation of the amendment at the respective unit.
2. Complete the Unit 1 cold leg and hot leg analyses at the increased sump water temperature and install any required pipe support modifications.	Enclosure 1 Section 4.1.5	Prior to implementation of the Unit 1 amendment.
3. Revise the Unit 2 operating procedures to instruct the operators to secure a train of the recirculation spray system if no service water is available for that train.	Enclosure 1 Section 4.1.5	Prior to implementation of the Unit 2 amendment.

COMMITMENT	REFERENCE	DUE DATE
4. Complete the analysis of the impact of the increased equipment qualification profile at each unit and replace or re-qualify equipment, as necessary.	Enclosure 1 Section 4.1.6	Prior to implementation of the amendment at the respective unit.