October 5, 2007

Mr. Peter P. Sena III
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: CHANGES TO THE RECIRCULATION SPRAY SYSTEM PUMP START SIGNAL DUE TO THE CONTAINMENT SUMP SCREEN MODIFICATION (TAC NO. MD4290)

Dear Mr. Sena:

The Commission has issued the enclosed Amendment No. 280 to Facility Operating License No. DPR-66 for Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 9, 2007, as supplemented by letters dated August 8, August 23, and September 13, 2007.

The amendment addresses Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," by implementing TS changes that reflect the use of a new recirculation spray system pump start signal due to a modification to the containment sump screens, and replace the use of the LOCTIC code with the Modular Accident Analysis Program-Design Basis Accident (MAAP-DBA) code to calculate containment pressure, temperature, and condensation rates for input to the SWNAUA code, which ultimately changes the aerosol removal coefficients used in dose consequence analysis.

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

Sincerely,

/RA/

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures:
1. Amendment No. 280 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

October 5, 2007

Mr. Peter P. Sena III
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: CHANGES TO THE RECIRCULATION SPRAY SYSTEM PUMP START SIGNAL DUE TO THE CONTAINMENT SUMP SCREEN MODIFICATION (TAC NO. MD4290)

Dear Mr. Sena:

The Commission has issued the enclosed Amendment No. 280 to Facility Operating License No. DPR-66 for Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 9, 2007, as supplemented by letters dated August 8, August 23, and September 13, 2007.

The amendment will address Generic Safety Issue 191 “Assessment of Debris Accumulation on PWR [Pressurized Water Reactor] Sump Performance” by implementing TS changes that reflect the use of a new recirculation spray system pump start signal due to a modification to the containment sump screens and replace the use of LOCTIC with the Modular Accident Analysis Program-Design Basis Accident (MAAP-DBA) code to calculate containment pressure, temperature, and condensation rates for input to the SWNAUA code, which ultimately changes the aerosol removal coefficients used in dose consequence analysis.

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

Sincerely,

/RA/

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures:
1. Amendment No. 280 to DPR-66
2. Safety Evaluation

cc w/encls: See next page

ADAMS Accession No.: ML072680397 *Input provided. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	DRA/AADB/BC	DSS/SSIB/BC	DE/EICB/BC	DIRS/IOLB/BC
NAME	NMorgan	SLittle	MHart*	MScott*	WKemper*	NSalgado
DATE	9/29/07	9/26/07	08/29/2007	09/11/2007	09/11/2007	09/13/2007

OFFICE	DSS/SCVB/BC	DE/EEEB/BC	DE/EMCB/BC	DIRS/ITSB/BC	OGC/ NLO	LPLI-1/BC
NAME	RDennig*	GWilson*	KManoly	TKobetz	STurk	DPickett for MKowal
DATE	09/18/2007	09/21/2007	9/22/07	10/1/07	10/4/07	10/5/07

OFFICIAL RECORD COPY

DATED: October 5, 2007

AMENDMENT NO. 280 TO FACILITY OPERATING LICENSE NO. DPR-66 BVPS-1

DISTRIBUTION:

PUBLIC	RidsNrrDorlDpr	NPatel	WKemper
LPLI-1 R/F	RDennig	MScott	IAhmed
RidsNrrDorlLpl1-1	RLobel	MHart	RReyes-Maldonado
NMorgan	NKaripineni	NSalgado	ABoatright
RidsNrrLASLittle	GWilson	KMartin	RidsAcrsAcnwMailCenter
DWerkheiser, RI	RidsOGCMailCenter	KManoly	RidsRgn1MailCenter
CBasavaraju	ALewin	TKobetz	GHill (2)

cc: Plant mailing list

Beaver Valley Power Station, Unit Nos. 1 and 2

cc:

Joseph J. Hagan
Senior Vice President of Operations
and Chief Operating Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

James H. Lash
Senior Vice President of Operations
and Chief Operating Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Danny L. Pace
Senior Vice President, Fleet Engineering
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Jeannie M. Rinckel
Vice President, Fleet Oversight
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

David W. Jenkins, Attorney
FirstEnergy Corporation
Mail Stop A-GO-18
76 South Main Street
Akron, OH 44308

Manager, Fleet Licensing
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-2
76 South Main Street
Akron, OH 44333

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

Director, Fleet Regulatory Affairs
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-2
76 South Main Street
Akron, Ohio 44333

Manager, Site Regulatory Compliance
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-A
P.O. Box 4, Route 168
Shippingport, PA 15077

Richard Anderson
Vice President, Nuclear Support
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
Akron, Ohio 44308

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

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

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

cc:

Dr. Judith Johnsrud
Environmental Coalition on Nuclear Power
Sierra Club
433 Orlando Avenue
State College, PA 16803

Director
Bureau of Radiation Protection
Pennsylvania Department of
  Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

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

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

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 298
Shippingport, PA 15077

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 280
License No. DPR-66

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated February 9, 2007, as supplemented by letters dated August 8, August 23, and September 13, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 280, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to the first entry into Mode 4 coming out of 1R18, which begins September 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/ DPickett for

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: October 5, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 280

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.3.2-8	3.3.2-8
3.3.2-9	3.3.2-9
--------	3.3.2-9a
--------	3.3.2-9b
3.3.2-12	3.3.2-12
3.5.2-3	3.5.2-3
3.6.5-1	3.6.5-1
5.5-19	5.5-19

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 280 TO FACILITY OPERATING LICENSE NO. DPR-66

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

1.0 INTRODUCTION

By letter dated February 9, 2007 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML070440341), as supplemented by letters dated August 8 (ADAMS Accession No. ML072250025), August 23 (ADAMS Accession No. ML072390013), and September 13, 2007 (ADAMS Accession No. ML072600022), the FirstEnergy Nuclear Operating Company (FENOC, the licensee), submitted a request for changes to the Beaver Valley Power Station, Unit No. 1 (BVPS-1) Technical Specifications (TSs).

The supplemental letters dated August 8, August 23, and September 13, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 24, 2007 (72 FR 20383).

The proposed changes would address Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance" and Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors" by implementing TS changes that reflect the use of a new recirculation spray system (RSS) pump start methodology and changes in the method of calculating post-accident containment temperature, pressure, and condensation rates. Specifically, the proposed changes would revise the method for starting the inside and outside containment RSS pumps in response to a design-basis accident (DBA). The licensee has proposed to change the start signal for the RSS pumps from a fixed time delay to an Engineered Safety Feature Actuation System (ESFAS) signal based on a reactor water storage tank (RWST) Level Low, coincident with a Containment Pressure High-High signal. This change to the RSS pump start methodology results from the replacement of containment sump screens. Additionally, proposed changes to the calculation methodology used to determine aerosol removal coefficients for use in dose consequence analyses, specifically the design-basis loss-of-coolant accident (LOCA) analysis, would result in the use of the Modular Accident Analysis

Program-Design Basis Accident (MAAP-DBA) code instead of the LOCTIC code to calculate containment pressure, temperature and condensation rates for input to the SWNAUA code.

Specifically, the licensee proposes to amend the following TS sections to reflect implementation of the new RSS pump start methodology:

- TS 3.3.2, “ESFAS Instrumentation,” Table 3.3.2-1, Function 2, Containment Spray.
- TS 3.3.2, “ESFAS Instrumentation,” Table 3.3.2-1, Function 7, Automatic Switchover to Containment Sump.
- TS 3.5.2, “ECCS [emergency core cooling system]-Operating.”
- TS 3.6.5, “Containment Air Temperature.”
- TS 5.5.12, “Containment Leakage Rate Testing Program.”

2.0 REGULATORY EVALUATION

The General Design Criteria (GDC) included in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, did not become effective until May 21, 1971. The Construction Permit for BVPS-1 was issued in June 1970; consequently, this unit was not subject to GDC requirements (Ref. SECY-92-223, dated September 18, 1992). The updated final safety analysis report (UFSAR) states that BVPS-1 has been designed and constructed to comply with the “General Design Criteria for Nuclear Power Plant Construction,” published in July 1967 by the AEC. However, Appendix 1A of the UFSAR provides a discussion of the degree of conformance to the AEC GDC published as Appendix A to 10 CFR Part 50 in July 1971, which indicates that it meets the intent of the GDC.

- Criterion 16-*Containment design* which states, “Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.”

- Criterion 38-*Containment heat removal* which states, in part, “A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident, and maintain them at acceptably low levels.”

- Criterion 50-*Containment design basis* which states, “The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by [10 CFR] 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

In reviewing this license amendment request (LAR), the NRC staff used the following sections of NUREG-0800, "Standard Review Plan": 6.2.1, "Containment Functional Design;" 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments;" and 6.2.2, "Containment Heat Removal Systems."

The regulatory requirement for which the NRC staff based its review of the changes to the safety system instrumentation trip set point for the ESFAS automatic protective devices and the addition of an ESFAS function with necessary editorial and nomenclature changes to clearly identify and reflect the licensed design, on Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," which requires a nuclear power plant TSs to include, among other things, limiting conditions for operation (LCOs), surveillance requirements (SR), administrative controls, and limiting safety system settings for automatic protective devices.

The NRC staff evaluated the licensee's analysis of the radiological consequences of the postulated design-basis LOCA, after the described implementation of the proposed TS and DBA analysis methodology changes, against the dose criteria specified in 10 CFR 50.67, "Accident source term."

In its review, the NRC staff considered the Standard Review Plan (SRP) 15.0.1 and the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 and Table 6 of Regulatory Guide (RG) 1.183. The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulatory standards and guides:

- 10 CFR 50.67, "Accident source term."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room."
- SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms."
- SRP Section 6.4, "Control Room Habitability."
- SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System."
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

The regulatory dose acceptance criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR) for the duration of the accident, 25 rem TEDE at the exclusion area boundary (EAB) for any 2 hour period, and 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the accident. The dose acceptance criterion in the Emergency Response Facility (ERF) is 5 rem TEDE for the duration of the accident to show compliance with the regulatory requirements of NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50, and the BVPS-1 current licensing basis.

The licensee has proposed a change to SR 3.5.2.7, which covers the periodic inspection of the containment sump screen assembly and trash racks relied upon by the ECCS and containment spray system (CSS) for long-term functionality. The licensee's response letter to GL 2004-02 dated September 6, 2005 (ADAMS Accession No. ML0525100411) described the NRC's requirements regarding the long-term functionality of the ECCS and CSS that are applicable to BVPS-1. The regulatory requirements pertinent to the proposed TS change are summarized below.

- Paragraph (b)(5)("long-term cooling") of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," states that "after any calculated successful initial operation of the emergency core cooling system, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

- GDC 35 of Appendix A to Part 50, "Emergency Core Cooling," states, in part, that "a system to provide abundant emergency core cooling shall be provided" following a LOCA, and that suitable redundancy shall be provided to ensure the system safety functions can be accomplished assuming a single failure.

- GDC 41 of Appendix A to Part 50, "Containment atmosphere clean-up," states, in part, that "systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce...the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained."

The basis for PWR licensees to demonstrate compliance with the requirements of 10 CFR 50.46 above is documented in GL 2004-02. The primary purpose of GL 2004-02 was to request that PWR licensees evaluate the performance of their containment recirculation sumps and implement any modifications necessary to ensure compliance with applicable regulatory requirements on a mechanistic basis in light of the technical issues associated with GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." The GL requested that PWR licensees complete actions necessary for compliance with applicable regulatory requirements using the updated information associated with GSI-191 by December 31, 2007. Prior to this date, GL 2004-02 concluded that licensees' compliance with their current licensing bases was sufficient to support continued plant operation.

The regulatory requirements and guidance which the NRC staff applied to its review of the equipment qualification (EQ) analysis and emergency diesel generator (EDG) loading impact included the following:

- 10 CFR Part 50, Appendix A, GDC 17 requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems and components important to safety. The onsite system is required to have sufficient independence, redundancy and testability to perform its safety function, assuming a single failure, and the onsite system is required to be supplied by two independent circuits. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of, or coincident with, the loss of power from the unit, the offsite transmission network, or the onsite power supplies.

- 10 CFR 50.36, "Technical specifications," requires a licensee's TS to establish LCOs and SRs for equipment that is required for safe operation of the facility. Specifically, Section 50.36(c)(2)(ii) stipulates the LCO requirements.

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires licensees to establish programs to qualify electric equipment important to safety.

- RG 1.9, Revision 2, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants."

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's review covered changes to operator actions, human-system interfaces, procedures, and training needed for the proposed TS modifications. The NRC staff's evaluation for human factors was based on the following documents: GDC-19; 10 CFR 50.120; 10 CFR Part 55; American National Standards Institute and American Nuclear Society aquatic nuisance species Standard 58.8 1994, "Time Response Design Criteria for Safety-Related Operator Actions;" and the guidance in GL 82-33. Specific review criteria are contained in SRP 13.2.1, 13.2.2, 13.5.2.1, and Chapter 18.0.

3.0 TECHNICAL EVALUATION

3.1 Containment and Net Positive Suction Head (NPSH) Analysis

BVPS-1 is a three-loop Westinghouse PWR originally licensed with a subatmospheric containment design. However, by letter dated February 6, 2006 (ADAMS accession number ML060100325), the Commission approved conversion of the BVPS-1 containment from subatmospheric to atmospheric operating conditions. The engineered safeguards features that mitigate a LOCA or main steamline break accident (MSLB) event include the following (Chapter 6 of the UFSAR):

- A safety injection (SI) system that injects borated water into the cold legs of all three reactor coolant loops.

- Two separate low-head safety injection (LHSI) subsystems, either of which provides long-term removal of decay heat from the reactor core.

- Two separate subsystems of the spray system, quench spray subsystem (QSS) and RSS, which, when operating together, reduce the containment pressure and temperature, and remove heat from the containment. The RSS transfers the heat from the containment to the service water (SW) system.

The QSS consists of two pumps which start on a Containment Pressure High-High signal and draw suction from the RWST, and spray into containment via spray headers to reduce containment temperature and pressure. The RSS system consists of four pumps. Two RSS pumps are located inside the containment (inside recirculation spray (IRS)) and take suction directly from the containment sump, while the other two RSS pumps are located outside the containment (outside recirculation spray (ORS)) and take suction from the containment sump through a dedicated line. The RSS pumps currently start after a time delay following receipt of a Containment Pressure High-High signal. Each RSS train has a recirculation spray heat exchanger that is cooled by SW (on the tube side) for long-term containment heat removal. The safety injection system (SIS) consists of two of the three charging pumps (which perform the charging functions during normal operations) acting as high-head safety injection (HHSI) pumps, and two LHSI pumps. The SI pumps take suction from the RWST and inject into the reactor coolant system cold-legs until RWST reaches its low-low level point, at which time transfer to recirculation mode is initiated. The LHSI pumps automatically realign to take suction from the containment sump and supply flow to HHSI pump suctions. Currently, the RSS pumps start after a short time delay following receipt of a containment pressure high-high signal. The pumps start after a delay of approximately 300 seconds. At this start time, there is limited quantity of water in the containment sump for ECCS recirculation. However, there is adequate net positive suction head (NPSH) margin for the pumps to support the current licensing basis for operability of the RSS and SIS pumps.

Because the RSS and SIS system pumps use suction from the containment sump, the resolution of NRC GL 2004-02 affects the operation of IRS, ORS and LHSI pumps. Appendix A to RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Revision 3, dated November 2003, gives different criteria for showing adequate pump performance depending on whether the sump strainer is fully or partially submerged when the LHSI and RS pumps are operating. For a fully submerged strainer, the strainer debris head loss must be less than or equal to the NPSH margin. For a partially submerged strainer, the strainer debris head loss must be less than one-half the pool height (the Nuclear Energy Institute (NEI) report, NEI-04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," also recommends the same criteria). The adoption of a licensing basis consistent with GL 2004-02 will result in an increase in the containment sump strainer head loss. Therefore, it is necessary to increase the static height of water in the containment sump to increase the NPSH margin available. The licensee proposed to accomplish this by changing the start signal of the RSS pumps from a fixed time delay from a timer, to an ESFAS signal based on a RWST low level coincident with a Containment Pressure High-High signal. Starting the RSS pumps on this coincident signal provides 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.

3.1.1 Containment Analysis

The current BVPS-1 licensing basis analysis methodology for LOCA containment response is the Modular Accident Analysis Program-Design Basis Accident (MAAP-DBA). The methodology change to replace LOCTIC with the current MAAP-DBA code was approved by letter dated February 6, 2006 (ADAMS accession number ML060100325).

The licensee performed calculations using the MAAP-DBA code to determine the impact of the proposed change to RSS pump start on the containment pressure and temperature response.

The licensee stated that a complete set of break sizes and locations was analyzed, including large-and small-break LOCAs and the spectrum of MSLB. The licensee stated that in addition to input changes to reflect the strainer heat sink metal mass and surface area, and RSS pump start signal, other inputs were modified to increase operating margin. For instance, the QSS and RSS minimum performance curves were reduced to allow for more operating margin for minimum pump performance.

The licensee stated that a LOCA results in the maximum peak containment pressure. However, the peak containment pressure remains below the design limit of 45 pounds square inch gauge (psig). The licensee further stated that the maximum containment pressure decreased from the current licensing basis. The maximum calculated value changed from 43.3 to 43.1 psig. In reference to the MSLB, the licensee stated that peak containment pressure remained the same. The results remain bounded by the peak containment pressure resulting from a LOCA.

With regards to temperature, the licensee stated that peak containment temperature occurs following an MSLB. The calculated maximum containment temperature is 355.9 °F. The maximum containment temperature is slightly above (< 1°F) the current Equipment Qualification (EQ) envelope. The longer term EQ temperature profile is governed by the LOCA, which is expected to increase due to the proposed delay in starting the RSS pumps. The licensee stated that the accident temperature profile for LOCA slightly exceeds (< 2°F) the existing EQ profile for a brief period at approximately 3 hours following accident initiation.

The licensee stated that the slight increase in the maximum and long-term containment temperatures is due to an increase in the maximum initial containment temperature from 105 °F to 108 °F. In response to the NRC staff's letter dated July 3, 2007 (ADAMS accession number ML071640121), the licensee stated that the purpose of increasing the maximum initial containment temperature is to provide additional operating margin. The ventilation system consists of three Containment Air Recirculation (CAR) fans. Normally, two of three fans are in service and the third is in standby. During the hottest months of the year, all three fans are required to maintain the containment temperature within limits, resulting in no standby unit availability. The licensee stated that the higher containment temperature analysis limit was intended to minimize the amount of time when all three fans are required. The licensee further stated that following the steam generator replacement at BVPS-1 in the spring of 2006, the containment temperatures had increased slightly, however they did not exceed 105 °F. Evaluations to determine the impact of the steam generator replacement and the extended power uprate demonstrated that the containment ventilation system is capable of handling the anticipated increase in heat load to maintain the containment at or below 108 °F.

The maximum containment liner temperature increased from 254.1 °F to 257.9 °F. However, the liner temperatures for all the cases analyzed met the acceptance limit of 280 °F (Tables D-5 and D-6 of Attachment D to Reference 1).

3.1.2 NPSH Analysis

The licensee performed calculations 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 pumps. The calculations establish the margin in available NPSH without taking into consideration the containment sump strainer head loss. In accordance with GL 2004-02, the licensee proposed to compare the margin with the head loss across the containment sump strainer following completion of strainer testing with the plant-specific debris loading. Attachment D and Section 4.1.2 of the enclosure of Reference 1 in the licensee's letter dated February 9, 2007, described the NPSH analysis and the results of the calculations.

In the current analysis, containment overpressure is credited in calculating the available NPSH for both the RSS pumps and LHSI pump when taking suction from the containment sump.

The licensee reported a reduction in the available NPSH for the IRS pump and an increase in the available NPSH for the ORS pump. The licensee attributed this to a more accurate modeling of the direct QSS flow to the RSS pumps (Section 4.1.2 of the enclosure to Reference 1). 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, whereas the current analysis assumed the enhancement flow to be equally divided between the ORS and IRS pumps. The licensee stated that the revised arrangement compensates for the higher suction piping loss in the ORS pump. The licensee explained that the current analysis made a simplification of the flow distribution from the QSS system to the ORS and IRS pumps, whereas in the revised model, a more detailed approach with system hydraulic flow models are used to calculate the total flow and distribution under various conditions of RWST level, containment pressure, pump performance, etc. The licensee further stated that restriction orifices in the system control the flow distribution and the revised approach does not compromise the flow distribution.

The licensee reported that the available NPSH for the IRS pumps decreased from 17.6 to 14.8 feet, but for the ORS pumps, the available NPSH increased from 11.8 to 14.1 feet. As previously noted, the available NPSH in the modified analysis does not include a strainer head loss component. The revised NPSH for all the RSS pumps is 9.8 feet, which results in a margin of 5.0 feet in available NPSH for the IRS pumps and 4.3 feet for the ORS pumps. The minimum water level above the bottom of the sump increased from 1.25 to 4 feet for the limiting case (typically a smaller break).

The LHSI pumps re-align to take flow from the containment sump in the recirculation mode of SI with transfer to recirculation occurring after the RSS pump start based on the transfer setpoint. The available NPSH for the LHSI pump will be reduced from 27.2 to 16.2 feet. The required NPSH is 10.6 feet and, therefore, the NPSH margin is 5.6 feet. The combined results of the RSS and LHSI pump NPSH calculations establish an upper limit of 4.3 feet (IRS pumps) for design of the new containment sump strainer in terms of available margin for head loss across the debris laden strainer.

3.1.3 Summary of Containment and NPSH Analysis

The results of the containment analysis using MAAP-DBA, as shown in Tables D-1 and D-2 (Reference 1), indicate a small reduction (0.1 to 0.2 psig) in the maximum containment pressure. The maximum containment pressure is governed by LOCA. The licensee stated that the LOCA peak pressure is not affected by the proposed change to the RSS pump start signal or timing, since the peak pressure occurs prior to operation of any CSS and that the small reduction in pressure is due to the heat sink provided by the new containment sump strainer. Tables D-1 and D-2 also show that the time required for containment pressure to be reduced to less than 50% of the peak pressure is considerably less than the acceptance criteria of 24 hours. Tables D-3 and D-4 (Reference 1) also show that containment peak pressure resulting from an MSLB is bounded by the LOCA results. The MSLB analysis does not credit the operation of the RSS pumps, and therefore theoretically, should be unaffected by the proposed change in the start signal. The actual results show that peak pressure due to MSLB increased slightly, which the licensee attributed to input changes to the QSS pump flow.

The containment temperature profiles for the limiting LOCA and MSLB cases are shown in Figures D-5 and D-6 (Reference 1). The peak temperature (governed by MSLB) increased by less than 1 °F above the current EQ profiles. For the longer term post accident temperatures (governed by LOCA), the results show an increase of less than 2 °F above the current EQ profile for a short duration of time. The licensee indicated that the primary reason for the slight increase in temperature is due to an increase in the initial containment temperature from 105 °F to 108 °F. The licensee explained the reasons for assuming a higher initial temperature and provided adequate basis in support of the assumption. The maximum calculated liner temperatures are shown in Tables D-5 and D-6 (Reference 1). All containment liner temperature results meet the acceptance limit of 280 °F.

The limiting NPSH results for the RSS and LHSI pumps are shown in Table D-9 (Reference 3). The NPSH transient results for the most limiting cases are shown in Figure D-7 (Reference 3). Based on the results provided by the licensee, the minimum NPSH margin for the pumps is 4.3 feet for RSS pumps and 5.6 feet for the LHSI pump. The containment overpressure is credited in calculating available NPSH for both RSS pumps and LHSI pumps. The licensee has stated that consistent with RG 1.82 requirements, parametric studies have been performed to bias inputs to minimize containment pressure while maximizing sump water temperature (and vapor pressure). The licensee made a commitment, per GL 2004-02, to compare the available NPSH margin for the RSS pumps with the head loss across the containment sump strainer following completion of strainer testing with the plant specific debris loading. Based on the licensee's usage of the NRC staff approved methodology (MAAP-DBA), compliance with the applicable RG and SRP, and the commitment to compare the available NPSH margin with the strainer head loss following completion of testing, the NRC staff finds the NPSH analysis acceptable.

3.1.4 TS Changes

The proposed TS changes to TS 3.3.2, reflecting the changes to the RSS pump start signal are appropriately modeled and reflected in the licensee's submittals. The licensee provided reasons and the basis for the proposed change to TS 3.6.5 to increase the upper limit on containment air temperature. The change was appropriately included in the licensee's containment analysis. The proposed TS change to TS 5.5.12 appropriately incorporated the calculated containment internal pressure from the revised containment analysis. Therefore, the NRC staff finds the above referenced TS changes, as described in the References 1 and 3, are acceptable with respect to the containment and NPSH consequences of DBA.

3.2 Containment Spray System ESFAS Instrumentation

The current TS ESFAS instrumentation LCO Table 3.3.2-1 does not indicate that the Containment Spray function is provided by two spray systems; Quench Spray and Recirculation Spray. The current TS table indicates that the Containment Spray function is initiated either manually, by Automatic Actuation Logic and Actuation Relays, or by Containment Pressure-High High signals. The proposed changes divide the Containment Spray system function into

Quench Spray and Recirculation Spray functions; each with its own initiation signals, and the respective requirements for applicable modes, required channels, operating conditions, SRs, and the setpoint allowable values. Containment Spray system initiation signals and the LCO requirements in the current TSs will become applicable to the Quench Spray function. The Recirculation Spray function, with its initiation signals, and the LCO requirements will be added in the proposed TSs.

The bases for these TS changes were provided in the LAR, which justified each of the initiation signals and their instrumentation LCO requirements. Recirculation spray will be initiated by Automatic Actuation Logic and also by RWST Level Low coincident with Containment Pressure High-High signals. The licensee proposed allowable values (AVs) for both RWST level and containment pressure instrumentation setpoints. The RWST Level Low allowable value has both upper and lower limits. The AV 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 AV upper limit ensures adequate NPSH to the Recirculation Spray pumps. The licensee stated that the RWST Level Low and Containment Pressure High-High instrumentation setpoint uncertainties were developed in accordance with Westinghouse Setpoint Methodology for Protection Systems, WCAP-11419, Rev. 5 for BVPS-1, which the NRC staff has previously reviewed and approved as documented in the NRC staff's safety evaluation dated January 11, 2006 (ADAMS accession number ML053530143). By letter dated August 8, 2007, the licensee provided a sample setpoint calculation which the NRC staff has reviewed and found acceptable.

The NRC staff found that the proposed applicable modes, required number of instrumentation channels, operating conditions, and the SRs for each of the three instrumentation signals for the start of RSS pump are acceptable. The NRC staff also found that the proposed instrumentation setpoint AVs for the RWST Level Low and Containment Pressure High-High instrumentation, to automatically start Recirculation Spray of the containment, are acceptable.

3.3 Radiological Consequences of the Design-Basis LOCA

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed LAR, as they relate to the radiological consequences of the design-basis LOCA. Information regarding this analysis was provided in Attachment E to the Enclosure of Reference 1. The licensee's evaluation verified that the offsite, control room, and emergency response facility (ESF) dose consequences remained within limits after implementation of the proposed changes. The current design-basis LOCA analysis was reviewed and approved by the NRC staff, by letter dated September 10, 2003 (ADAMS accession number ML032530204); this current design-basis LOCA analysis remains largely unchanged by this LAR. Therefore, the NRC staff's review is limited to the changes to the existing design-basis LOCA analysis, which includes all new assumptions, inputs, and methods used by the licensee to assess the impacts of the LAR. Where necessary, the NRC staff performed independent calculations to confirm the conservatism of the licensee's analyses.

To perform independent confirmatory dose calculations, the staff used the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604, "A Simplified Model of Aerosol Removal by Containment Sprays," dated June 1993. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and the resulting radiological consequences at selected receptors.

3.3.1 Loss-of-Coolant Accident (LOCA)

The proposed TS changes will result in a modification of the post-LOCA RSS pump start methodology. The licensee has proposed to change the start signal for the RSS pumps from a fixed time delay from to an ESFAS signal based on an RWST Level Low, coincident with a Containment Pressure High-High signal. The licensee believes that 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 containment. The licensee also stated that 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.

The RSS helps mitigates activity release and draws water from the RWST, as credited in the current design-basis LOCA analysis. Therefore, with respect to evaluating LOCA dose consequences, the implications of this RSS pump start methodology change affects spray removal of aerosol iodine released into containment following the postulated LOCA, as well as the timing of activity releases from the RWST leakage pathway. This proposed method for starting RSS pumps will delay activation of the RSS, which will result in a short-term increase in air leakage from the containment and a short-term reduction in spray removal of airborne radioactivity from the containment atmosphere.

To reflect the implementation of the new RSS pump start methodology, the licensee proposed changes to the current design-basis LOCA dose consequence analysis. All changes to the input parameters used in calculating the radiological consequences of the DBA come as a result of the licensee implementing the use of a new code to determine post-accident containment conditions. Currently, to determine the containment diffusiophoresis-based iodine activity removal using the SWNAUA code, the input parameters of containment pressure, temperature, relative humidity and steam condensation rates are calculated using the LOCTIC code along with a NUREG-1465-based delayed ECCS scenario. The acceptability of this methodology is documented in the NRC staff's letter dated September 10, 2003. For this LAR, the licensee used the MAAP-DBA code, instead of LOCTIC, to calculate containment pressure, temperature, and condensation rates for input to the SWNAUA code. Thus, the EPRI code, MAAP-DBA, is the licensing basis code being used for BVPS-1 containment pressure and temperature calculations. When compared to the LOCTIC code calculations, the MAAP-DBA code approach is consistent with that used in LOCTIC and also calculates conservative results for containment pressure, temperature, and condensation rates.

Table 3.1 identifies the input parameter changes resulting from this implementation of the MAAP-DBA code to calculate a new containment and coolant response that reflects modified RSS pump start time.

Table 3.1
Comparison of Parameter Changes from Current to Proposed Licensing Basis

Input Parameter	Current DBA Value	Proposed DBA Value
RSS Spray Period		
Start (seconds)	720	85.4
End (days)	4	4
Aerosol Spray Removal Rates		
Containment Sprayed Region	Figure 5.3.6-1*	Figure 3.1**
Containment Unsprayed Region	Figure 5.3.6-2*	Figure 3.2**
RWST Backleakage		
Leakage Onset (seconds)	2186	1782
Activity Venting Onset (seconds)	5178	3055
Release Fractions	Figure 5.3.6-3*	Figure 3.3**
ESF Leakage		
Start (seconds)	300	1200
End (days)	30	30
Minimum Sump Volume		
5 minutes – 30 minutes (ft^3)	9800	19,111
30 minutes – 2 hours (ft^3)	28,600	25,333
2 hours – 30 days (ft^3)	65,600	43,577

* Letter dated June 5, 2002 (ADAMS accession number ML021620298)
** Reference 1

Figures E-1 and E-2, from Attachment E of the Enclosure to Reference 1, illustrate the resulting iodine removal rate profile in containment, as calculated with the SWNAUA code using the new input parameters determined by MAAP-DBA. Figure E-3, from Attachment E of the Enclosure to Reference 1, illustrates the change in the RWST iodine release fraction resulting from the RSS pump start methodology change and subsequent containment response recalculation using MAAP-DBA.

No compensatory changes were made to the design-basis LOCA analysis by the licensee to reduce the dose consequences associated with the conservative changes detailed in this LAR. Also, the licensee has not made any methodological changes to the calculation of airborne iodine activity removal in containment or to the release of activity from the RWST. As shown in Table 3.2, the newly calculated dose consequences still remain well within the regulatory limits.

Table 3.2
Licensee Calculated DBA LOCA Radiological Dose Consequences

	Control Room	Emergency Response Facility	EAB	LPZ
	TEDE (rem)	TEDE (rem)	TEDE (rem)	TEDE (rem)
Current	2.0	3.0	14.0	2.5
Recalculated	2.5	3.5	16.5	3.0
Regulatory Criterion	*5*	*5*	*25*	*25*

The NRC staff performed an independent calculation using the licensee's input and approximating the licensee's revised iodine removal, and confirmed that the licensee's analysis results are acceptable. Therefore, with respect to DBA radiological dose consequence analysis, the NRC staff finds the licensee's proposed changes to the LOCA dose analysis acceptable.

3.3.2 Atmospheric Dispersion

The licensee used atmospheric dispersion factors ($^{\chi}/Q$ values) from their current licensing basis. For the CR, EAB, and LPZ dose estimates, the $^{\chi}/Q$ values implemented in the LAR were previously reviewed and approved by the NRC staff by letter dated September 10, 2003.

3.3.3 Summary of Revised LOCA Analysis

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the CR, EAB, and LPZ radiological consequences are within the dose acceptance criteria provided in 10 CFR 50.67 and the accident dose guidelines specified in SRP 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance. The assumptions found acceptable to the NRC staff were discussed in Section 3.3.1. As needed, the NRC staff performed independent assessments of radiological dose consequences using the licensee's assumptions and confirmed the licensee's results. The NRC staff finds that the CR, EAB, and LPZ dose consequences, estimated by the licensee for the LOCA, as presented in Table 3.2, meet the dose acceptance criteria in 10 CFR 50.67 of 25 rem TEDE at the EAB and LPZ, and 5 rem TEDE in the CR, and are, therefore, acceptable. The NRC staff also finds that the LOCA dose consequence estimated by the licensee in the ERF, as presented in Table 3.2, meets the regulatory criteria of NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 of 5 rem TEDE; this result is acceptable as well.

3.3.4 TS Changes

The proposed TS changes are appropriately modeled and reflected in the licensee's submitted dose analysis. Therefore, with respect to the dose consequences of the design-basis LOCA, the NRC staff finds that the proposed TS changes, as described in Reference 1, are acceptable with respect to the radiological consequences of DBAs.

3.4 Containment Sump SR

3.4.1 System and Design Description

The recirculation spray pumps take suction from the containment sump, which is enclosed by a protective screen assembly. The assembly consists of three sections, the inclined bars and two stages of screening. The inclined bars act as trash screens to prevent large pieces of debris from reaching the roughing and final screens. After the bars, there are two stages of screening, the first consisting of a coarse mesh and the second a fine mesh with an opening slightly smaller than the size of the smallest nozzle orifice in the recirculation spray header. The assembly is divided at the center line by screening so that failure of either half does not adversely affect the other half.

A new multi-strainer installation is planned for BVPS-1 in the fall 2007 refueling outage. The strainer is designed with multiple pockets. The new strainer is fabricated from perforated stainless steel plate. The perforated plate has a 3/32 inch opening size, which is slightly smaller than the size of the smallest coolant passage in the reactor core and the orifice size of the spray nozzles. The strainer design incorporates a grating structure covered in solid plate to prevent damage to the strainer top-hat assemblies. The licensee stated that the replacement strainers do not require a vertical outer trash rack to protect the primary strainer surface.

3.4.2 Proposed TS Change

Currently, TS SR 3.5.2.7 reads as follows:

> Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.

The licensee has proposed the following revision to SR 3.5.2.7:

> Verify, by visual inspection, that accessible regions of the ECCS containment sump suction inlet are not restricted by debris and that the accessible regions of the strainers show no evidence of structural distress or abnormal corrosion.

The licensee indicated that the proposed revisions would not fundamentally alter the current inspection practice required by SR 3.5.2.7. Specifically, the licensee will continue to be required to visually inspect the containment sump suction inlet to verify that it is not restricted by debris and that its debris filters show no evidence of structural distress or abnormal corrosion.

3.4.3 Licensee Justification for Proposed TS Change

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," and adding the wording "accessible regions" to describe the extent of the visual inspection.

The licensee indicated that the replacement of "trash racks and screens" with "strainers" in SR 3.5.2.7 provides a more appropriate description of the sump configuration after the installation of a larger strainer assembly to address GL 2004-02 is completed. The licensee considers the removal of the word "train" to be a clarification of the current SR because the sump strainer is a combined header for both ECCS trains, and the word "train" is not needed. The addition of the words "accessible regions" to the SR is due to the size, complexity and location of the new containment sump strainers that limit the extent of the visual inspection. 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.

3.4.4 NRC Staff's Evaluation

In determining the adequacy of the licensee's proposed TS change, the NRC staff considered whether the planned replacement strainer assembly is capable of fulfilling the design functions of the existing screen and trash rack configuration under the current licensing basis. The design basis function of the emergency sump is to provide a long-term water source for the recirculation function of the ECCS and RSS. The containment depressurization system is designed in accordance with RG 1.82, Rev. 0, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," as it relates to the design of sumps for ECCS and CSS.

3.4.4.1 Changing of the Terminology "trash racks and screens" to "strainers"

As described in the BVPS-1 UFSAR, the function of the trash rack is to prevent large debris from reaching the fine inner screen. This debris could have a significant impact on the current strainer with a comparatively low strainer-to-debris surface area ratio. The replacement strainers have more than 22 times the surface area of the old strainers, increasing the strainer-to-debris surface area ratio. The licensee stated that the complex strainer geometry provides a variation in surface contour such that large debris will not completely impede flow. BVPS-1 concluded that the large size, strainer layout, and complex geometry would make it difficult for large debris to fully obstruct the strainers. The NRC staff agrees with this conclusion.

Regulatory Position C.6 in RG 1.82 states that the strength of the trash rack should be considered for protection of the fine inner screen from large debris. The licensee stated that the new sump strainers are robust components made of stainless steel perforated plate that cannot be punctured or cut by sharp debris. Therefore, the licensee concluded that the trash rack surrounding the replacement sump strainers is no longer required to protect the strainers from being punctured or cut.

The NRC staff concludes, based on the information provided by the licensee and the robust construction of the now strainers that trash racks are not needed to protect the strainers from impingement by large debris.

3.4.4.2 Removal of the word "train"

By letter dated August 8, 2007, the licensee provided clarification of whether the strainer replacement represents a change from two independent sumps to a shared sump. The BVPS-1 has a single recirculation sump. In response to an NRC staff question, the licensee provided a copy of its 10 CFR 50.59 evaluation for BVPS-1, entitled "Containment Sump Strainer," dated June 29, 2007. This evaluation notes that, the existing strainer has a screen separating one half of the strainer from the other, such that a failure of one strainer will not have an impact on the

RS and LHSI equipment downstream of the other strainer. The new strainer does not include such a screen. This has the advantage of making the entire strainer area available to support flow into either train.

There are benefits of having independent strainers as well. These include the ability to handle some beyond-design-basis situations. For example, excessive blockage could possibly be mitigated by stopping flow through one strainer. Independent strainers also provide some protection from a random beyond-design-basis passive failure. While having two sumps clearly supports demonstration of compliance with GDC 35, the NRC staff accepts that single or common sumps are also compliant with the GDC, if there is reasonable assurance that single failure of the strainer will not occur. The NRC staff, in reviewing submittals related to GSI-191, does not believe there is an overriding case for one configuration or the other. Licensees have a number of options for addressing the issue, as long as compliance with the regulations is demonstrated.

The licensee's 10 CFR 50.59 evaluation stated that the strainer is ruggedly designed, constructed of corrosion-resistant stainless steel, not susceptible to potential jet impingement or pipe whip, and not exposed to nearby nonseismic components. The licensee concluded that there is no mechanism or event available to jeopardize the structural integrity of the strainer and that a passive failure of the strainer is not credible.

As discussed, the licensee has implemented periodic inspections to ensure that accessible regions of the new strainer are free of structural distress, abnormal corrosion, and debris. The licensee has described design features (metal covers) over the inaccessible regions of the strainer that the licensee concludes will preclude structural distress to those regions. These measures are consistent with standard industry practices. Further, the NRC staff finds that these measures, if properly implemented, are adequate to detect strainer damage that could adversely impact the ability of the ECCS to perform as designed post-LOCA. Therefore, the NRC staff finds the proposed change to the TS regarding removal of the word "train" to be acceptable from the perspective of potential for undetected latent damage leading to strainer failure.

3.4.4.3 Adding the words "accessible regions"

In Reference 2, the licensee clarified the scope of SR 3.5.2.7. The licensee stated that the accessible regions of the strainers will be inspected each outage for evidence of structural distress, abnormal corrosion, and debris, as required by the proposed TS. The inaccessible regions of the strainers are protected by diamond checker plates over the strainer. The licensee stated that there is no mechanism to cause structural distress or corrosion in the inaccessible regions that would not be present in the accessible regions. The protective cover keeps out any mechanism that could cause structural distress. Because the inaccessible regions are in close proximity to the accessible regions and share the same materials of construction and environment, the licensee concluded that the absence of abnormal corrosion on accessible portions of the strainer provides reasonable assurance of similar conditions in inaccessible regions. The licensee stated that there is no mechanism for debris introduction to the inaccessible regions other than the possibility of debris introduction during testing of the RSS pumps. The strainers are outside of the test pool.

Based upon the information described above, the NRC staff considers the replacement strainer configuration as meeting the intent of the current sump performance licensing basis because the filtration capacity associated with the replacement strainers' large, complex surface is significantly in excess of the filtration capacity associated with the existing screen. Under the current licensing basis, BVPS-1 demonstrates adequate sump functionality based on an assumption, from RG 1.82, Revision 0, that half the area of each of the existing sump screens is covered with debris such that water cannot flow through the blocked portion of the screen, while the other 50% is assumed to remain completely unblocked. Therefore, the NRC staff considers the replacement strainers to be functionally equivalent to (or better than) the existing screens under the non-mechanistic current licensing basis for satisfying the requirements of 10 CFR 50.46(b)(5) for long-term reactor core cooling. Therefore, the NRC staff considers the licensee's proposed TS SR change to be acceptable for the period prior to December 31, 2007.

Consistent with the intent of GL 2004-02, current licensing basis compliance is sufficient until December 31, 2007. No later than this date, the NRC staff has requested that licensees complete modifications to their licensing bases for containment recirculation sump performance to ensure consistency with the mechanistic methodology associated with GSI-191. Assurance that the licensee's replacement strainer design is adequate for satisfying the intent of GL 2004-02 will be reviewed by the NRC staff as part of its regulatory activities regarding GL 2004-02 and GSI-191, including reviews of licensees' supplemental responses to GL 2004-02, sample audits of licensees' sump performance calculations, and reviews of generic industry guidance and practices.

3.4.4.4.1 Pipe Whip, Jet Impingement, and Missile Impact

The NRC staff review focused on whether the planned replacement strainer evaluation has adequately considered potential dynamic effects of jet impingement, missile impact, and pipe whip. The NRC staff requested that the licensee provide additional information to ensure the structural integrity of the new passive strainer assemblies. The licensee provided a sketch showing the general arrangement of BVPS-1 containment sump strainer assembly. The new sump strainer will be located at elevation (El) 692-11” of the containment; on the bottom floor of the containment and entirely outside of the crane wall. High energy systems such as Feedwater, Main Steam, Steam Generator Blowdown and Reactor Coolant piping, are isolated from the sump by major structural features, such as walls and floors. The new containment sump strainer will be located adjacent to the containment liner at El 692’-11”. The BVPS-1 design is such that the polar crane wall serves as a barrier between the reactor coolant loops and the containment liner. In addition, the refueling cavity walls, various structural beams, the operating floor, and the crane wall, enclose each reactor coolant loop into a separate compartment, thereby preventing an accident which may occur in any loop from affecting another loop or the containment liner. The portions of the steam and feedwater lines within the containment have been routed behind barriers which separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces. This protection from the dynamic effects of pipe breaks is included in Section 5.2.6.3 of the BVPS-1 UFSAR. The existing compartments and the crane wall provide protection from high energy line break effects for the new containment sump strainer assembly.

Components which are considered to have a potential for missile generation inside the reactor containment are the following:

- Control rod drive shaft, and the drive shaft and drive mechanism latched together
- Certain valves
- Temperature and pressure assemblies.

Due to the location and existing protection, a missile from the control rod drive shaft cannot impact the new strainer.

The design and licensing basis for valve stems as potential missiles include only those valves in the region where the pressurizer extends above the operating deck. Valves in this region are the pressurizer safety valves, the motor operated isolation valves in the relief line, the air-operated relief valves, and the air-operated spray valves. Due to their location, these valve stems cannot impact the new strainer assembly.

Temperature elements are installed on the reactor coolant pumps, close to the radial bearing assembly. Based on the locations of these assemblies, a postulated missile cannot impact the new strainer.

The new BVPS-1 containment sump strainer design did not result in piping rerouting in the vicinity of the sump strainer.

3.4.4.4.2 ASME Codes

In response to the NRC staff's questions regarding the codes utilized in the structural design of the sump replacement strainer, the licensee provided the following information. The existing containment sump screen assembly consists of three sections: inclined grating and two stages of screening. The superstructure that supports the grating and screening is a platform that also provides support for the IRS pumps, the normal sump pumps and associated piping as well as other appurtenances. Portions of the existing sump screen assembly, which consists of the majority of the grating, and the screening, will be removed and replaced with a new containment sump strainer. The new containment sump strainer assembly consists of three interconnected components: modules, a channel box, and a suction box. The modules, channel box (including their support frames), and suction box assemblies, are primarily made from plates or shells. The superstructure which provides component support functions will remain in place.

The licensee used subsection NF of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III 2004 Edition including 2005 Addenda for the design and fabrication of replacement components and systems. ASME III Subsection NF was selected since it provides appropriate guidance for the design of plate and shell type components of the new strainer, versus the post and beam type design used for the existing screens. The new strainer does not provide any support function for any pumps or piping. The superstructure used for the existing screens will continue to provide the pump and piping support. The LHSI & RS piping is designed to ANSI B31.1.0, 1967 Edition through June 30, 1971 Addenda, with no requirements for the use of ASME III Subsection NF for support design. There is no specific method of evaluation discussed within the UFSAR on the design of the containment sump strainer assembly in regards to any design feature. The new strainer components are designed to conform to the ASME III Subsection NF Code, however, neither the existing strainer nor the new strainer is designated as an ASME III component. A comparison of the allowable limit equations identified in the vendor analysis, which used the 2004 Edition including 2005 Addenda of the ASME III, Division 1, Subsection NF Code, was performed against the 1998 Edition. The comparison determined that there is no significant difference in any equation used, concluding that the results would not be affected by using the earlier Code Edition. The comparison of the analysis against the earlier Code will be documented in the design package as part of the normal modification process prior to operational acceptance of the modification. The NRC staff finds the codes utilized in the design of strainer assembly acceptable.

3.4.4.4.3 Loads and Load Combinations

In response to the NRC staff's questions regarding the loads, and load combinations utilized in establishing the structural integrity of the sump replacement strainer assemblies and the discharge piping, the licensee provided the following information.

Load Combinations and Load Definitions are provided in the table below, which are considered in the strainer structural analysis.

Load Combination No	Load Combinations
1	DL (pool dry)
2	DL + OBE (pool dry)
3	DL + SSE (pool dry)
4	DL + OBE (pool filled)
5	DL + SSE (pool filled)
6	DL + WD + OBE (pool filled) + DP
7	DL + WD + SSE (pool filled) + DP
8	DL + LL (pool dry)

Load Definitions:
DL - Weight of strainers and supporting structure
WD - Weight of debris
DP - Differential pressure
OBE - Operating Basis Earthquake
SSE - Safe Shutdown Earthquake
LL - Live Load

Hydrodynamic masses as well as loads due to sloshing are taken into account with the earthquake loads. The NRC staff finds that the load combinations follow the guidelines of the RG 1.82.

3.4.4.4.4 Temperature Effect

Sliding joints are provided between ducts and supports, permitting differential expansion of the steel structure and concrete floor resulting in no significant temperature stresses. The load resulting from temperature differential between the bottom (submerged in hot sump water) and the top of the strainer (exposed to cooler containment spray) does not cause significant thermal stresses because the strainers are free to move in the vertical direction.

3.4.4.4.5 Structural Integrity of the Strainer Components

The structural components of the Strainer Modules include Side Wall, Upper Cover Plate, Lower Cover Plate, Perforated Sheet, Support Structure, Duct Plate, Anchor Plate, Anchor Bolts, and Anchor Bolts – End Plate. The Channel Box consists of Connection Duct Plates, Suction duct, and Suction duct Anchor Bolts. The components of the Suction Box include Suction Box support Element, Anchor Plates, Back Side Plates, Front Side Plates, Top Plates, Anchor Bolts, Sheet, and Side Plate – Sheet.

The computed stresses in the various structural components of the strainer assembly are within their allowable stress limits. The margin for each strainer component was determined. Most components were determined to have significant margin, and every analyzed strainer component was determined to be within its allowable stress limits.

3.5 Equipment Qualification Analysis and Emergency Diesel Generator (EDG) Loading Impact

3.5.1 EQ Analysis

The licensee proposed to change the upper limit on containment average air temperature from 105 °F to 108 °F. The licensee stated, "This change incorporates the revised containment analysis upper limit on containment average air temperature." This resulted in a maximum containment temperature slightly above the current EQ envelope. On Page 3 of Attachment D to the letter dated February 9, 2007, it is stated that the purpose of raising the limit is to allow for an increase in the containment operating band.

Section 4.1.6 of the enclosure to Reference 1 provides a discussion of EQ analysis and states that analysis of the impact of the increased equipment qualification 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.

The NRC staff requested the licensee provide any impact on its EQ program due to increase in the normal ambient temperature and radiation doses due to the proposed LAR. By letters dated August 8, 2007, and September 13, 2007, the licensee provided the EQ analysis and calculation summary.

The licensee provided the BVPS-1 EQ calculation summary and the accident analysis temperature results for various LOCA and MSLB cases in support to this LAR. The licensee established a new composite temperature profile by bounding the accident analysis temperature overall peak and duration results. The licensee performed EQ evaluations for all EQ equipment inside containment using this composite EQ temperature profile. The licensee, in its letter dated September 13, 2007, stated that (1) the pressure profile did not change due to this LAR and (2) the areas that experienced radiation dose increase do not contain any EQ equipment and no area became a harsh environment as a result of the dose increase. Based on the above analysis and EQ calculations, the licensee concluded that EQ equipment both inside and outside containment remains qualified for the proposed LAR and it was not necessary to replace or re-qualify any equipment as a result of this LAR.

Based on its review of the licensee’s submittals, the NRC staff finds that the licensee’s EQ analysis is in compliance with 10 CFR 50.49 for BVPS-1.

3.5.2 EDG Loading

In Section 4.1.7, of the enclosure to the letter dated February 9, 2007, the licensee addressed EDG loading. It stated that with the proposed RSS pump start logic, the inside containment RSS pumps will be started immediately following receipt of an RWST Level low signal coincident with a Containment Pressure High-High signal. The outside containment RSS pumps will be started following a 15-second delay after receiving the coincident signal. The 15-second delay limits the starting load on the EDG and maintains staggered pump start timing. The maximum load on the EDG will not increase as a result of this modification and will now occur at a later time due to the delay in starting the RSS pumps.

The NRC staff requested the licensee to confirm that the revised loading sequence has been re-evaluated to verify that it meets RG 1.9, as it pertains to its load accepting capability. The licensee provided an evaluation of the impact on the EDG due to this LAR. The revised loading sequence has been re-evaluated to verify that it meets RG 1.9, as it pertains to its load accepting capability of RSS motor loads in conjunction with other Engineered Safety Features step loads.

The BVPS-1 EDGs have an UFSAR loading limit of 2745 KW. The RSS pumps were designed to start at 210 seconds and 225 seconds on each EDG using a timer after the receipt of a Containment Pressure High-High Signal. The maximum coincident loading occurs on the "A" EDG at 15 minutes (2713.6 KW with a margin of 31 KW) and on the "B" EDG at 30 minutes (2682.9 KW with a margin of 62 KW).

Based on its review of the licensee’s submittal, the NRC staff finds that the revised loading sequence for BVPS-1 EDGs is consistent with the guidance of RG 1.9 and the existing EDG loading capacity bounds the revised loading for the proposed design change. Based on this information, the NRC staff concludes that the proposed design change is acceptable.

3.6 <u>Human Factors</u>

3.6.1 Operator Actions Affected

The licensee requested TS changes related to the modification of the RSS pumps start signal initiation. The proposed chance requires a coincident Containment Pressure High-High/ RWST Level Low ESFAS signal instead of a fixed time delay signal. This results in the RSS pump to start automatically as driven by plant conditions. This section evaluates the modified or additional operator manual actions proposed for addition to the Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs), the effect on existing operator action times, and any impacts to the control room environment.

3.6.2 New and Modified Operator Actions in EOPs and AOPs

The licensee will add a new operator manual action to the BVPS-1 EOPs to direct the operators to shut down two of the four operating RSS pumps after reaching transfer to SI recirculation mode (RWST Level Extreme Low). Specifically, upon receipt of the RWST extreme low level signal:

- RSS pump discharge valves to the LHSI header open
- RSS spray header isolation valves close
- LHSI pumps stop when the RSS pump discharge valves are fully open
- HHSI pump suction valves from the LHSI header open
- HHSI suction valves to the RWST close
- LHSI pump suction isolation valves from the RWST close

No action is required in the event where only a single train is operating. This new action involves resetting the containment isolation phase B relay and shutting down the pumps using the pump control switch. The purpose of this action is to limit the maximum flow through the containment sump strainer. This helps to minimize the strainer head loss.

Currently, the EOPs instruct the operators to monitor pump operation for signs of cavitation. If the RSS pumps are cavitating, procedures provide instructions for the operators to shut the pumps down until adequate inventory in the sump is available as indicated by the sump level indicators.

The licensee process requires that LARs are reviewed to determine what procedure changes are required including operator actions in the EOPs or by reference to other operating procedures. The associated Engineering Change Packages (ECPs) have similar requirements in the form of a Design Interface Evaluation (DIE). Procedure changes associated with the proposed RSS pump start signal modification have been identified and are scheduled for completion prior to amendment implementation. Procedure changes are currently evaluated under 10 CFR 50.59 to ensure consistency with the UFSAR and verify that the procedure changes can be made without NRC approval.

The NRC staff has reviewed the proposed new manual actions for the EOPs to support the LAR. The NRC staff determined that the new operator manual actions do not result in changes in either the operating or accident mitigation philosophies. Based on the licensee's description of the actions to be credited in the UFSAR, the licensee's commitment to using a tabletop approach to validate the procedure changes associated with the new operator actions and a technical verification performed on the changes as part of the validation process, the NRC staff finds the new operator manual actions acceptable.

3.6.3 Impact on Existing Operator Action Times

In response to the NRC staff's request for additional information (RAI), the licensee stated that operator response times will remain unaffected by the proposed LAR. The RSS pumps will start automatically upon receipt of the RWST level low signal as long as a Containment Pressure High-High signal is present.

The new operator action for EOPs needs to be completed before the RWST level reaches the Extreme Low level setpoint, which requires a minimum time of approximately 30 minutes to reach this setpoint. Currently a continuous action in EOP E-1, "Loss of Reactor or Secondary Coolant", directs the operator to go to procedure ES-1.3, "Transfer to Cold Leg Recirculation," when the RWST level is less than 15 feet. The new action of shutting down RSS pumps if more than two are operating will be added to ES-1.3. The licensee stated these new steps will not impact the automatic transfer functions or subsequent operator verification activities in ES-1.3 and will be completed in the 30 minutes allotted in the accident scenario.

3.6.4 Changes to Environments in which the Operators Complete Actions

The licensee stated that starting the RSS pumps on a coincident Containment Pressure High-High/RWST Level Low Engineered ESFAS signal would not involve a new operator manual action due to the RSS pump start function being automatic. The automatic function of the RSS pump start signal will not affect the operators' ability to stop the QSS pumps upon receipt of the RWST alarm, which indicates that the RWST is nearing empty. The timing of RWST drawdown is not impacted by the change in the start time of the RSS pumps since these pumps do not draw from the RWST. This action takes place in the control room and the environment will be unchanged.

Based on preliminary dose assessments conducted by the licensee, the NRC staff observed that the dose increases are minimal and are not expected to impact the ability of operators to safely complete the existing and new operator manual actions. The licensee also stated that the proposed RSS pump start modification should not adversely affect the habitability of the control room or result in additional heat concerns, smoke, toxic gases, effects of ventilation shutdown or impact the operator manual actions in the locations where the actions are to be taken and along access and egress routes.

3.6.5 Changes to Control Room Controls, Displays and Alarms

In response to the NRC staff's RAI, the licensee provided a list of changes to the control room including the addition of new status lights, updating annunciator windows, updating computer points, relocating status lights based on human factors, addition of status panel lights, and renaming status lights. These changes to the control room will notify the operators of the status of the RWST and other supporting equipment involved in the proposed LAR. The NRC staff reviewed the proposed changes to the control room and concluded that the changes reflect the equipment and logic changes made in the plant in support of the proposed LAR.

3.6.6 Changes to the Operator Training Program

In response to the NRC staff's RAI, the licensee stated that its training department completed a Design Interface Evaluation as part of the ECP program and identified Operations and Instrument and Controls (I&C) Maintenance as areas requiring training for both Units. Lesson plans will be updated to incorporate the appropriate subject matter. The training simulator will be modified and applicable training will be conducted for the required operations personnel. The training documentation has been identified for update and is being tracked as part of the ECP process. Existing operator training methods will not be affected by the proposed changes to the RSS pump start signal.

The NRC staff has reviewed the proposed changes to the operator training program. The NRC staff determines that the changes do not result in changes to the operating or accident mitigation philosophies.

4.0 LICENSEE COMMITMENT

The licensee stated that a new operator action will be incorporated as part of the overall sump modifications. The operator action is not directly required to support the proposed modification to the RSS start signal, but has been incorporated into the associated analysis.

A new operator action will be added to BVPS-1 EOPs to direct the operators to shut down two of the four operating pumps after reaching transfer to SI recirculation mode. No action is required in the event only a single train is operating. The licensee stated that this action will limit the maximum flow through the sump strainer, which helps to minimize the strainer head loss.

The licensee stated that the new operator actions will be validated during operator training prior to implementation.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 20383). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

8.0 REFERENCES

1. FirstEnergy Nuclear Operating Company letter L-07-017, "Beaver Valley Power Station, Unit Nos. 1 and 2, Licensing Amendment Requests Nos. 334 and 205," dated February 9, 2007.

2. FirstEnergy Nuclear Operating Company letter L-07-095, "Beaver Valley Power Station, Unit Nos. 1 and 2, Responses to a Request for Additional Information (RAI) dated July 3, 2007, in Support of Licensing Amendment Request Nos. 334 and 205 (TAC Nos. MD4290, MD4291," dated August 8, 2007.

3. FirstEnergy Nuclear Operating Company Letter L-07-105, "Beaver Valley Power Station, Unit Nos. 1 and 2, Supplemental Information for Licensing Amendment Request Nos. 334 and 205 (TAC Nos. MD4290 and MD4291)," dated August 23, 2007.

4. Technical Report NEI-04-07, Revision 0, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Volumes 1 and 2 (Safety Evaluation Report), December 2004.

5. NRC letter dated February 6, 2006, "Beaver Valley Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Issuance of Amendments Re: Containment Conversion from Subatmospheric Operating Conditions (TAC Nos. MC3394 and MC3395)," Amendment Nos. 271 and 153.

Principal Contributors: Nageswara Karipineni
Iqbal Ahmed
Aleem Boatright
Ruth Reyes-Maldonado
Chakrapani Basavaraju
Nitin Patel
Kamishan Martin

Date: October 5, 2007