October 12, 2006

Mr. David A. Christian
Senior Vice President
and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING IMPLEMENTATION OF GENERIC SAFETY ISSUE 191 (TAC NOS. MC9724 AND MC9725)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 250 to Renewed Facility Operating License No. DPR-32 and Amendment No. 249 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated January 31, 2006, as supplemented by letters dated February 23, June 21, and July 28, 2006.

These amendments revise the TSs to incorporate the changes to the operation of the containment, as discussed in Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactor," dated September 13, 2004.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Siva P. Lingam, Project Manager
Project Directorate II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 250 to DPR-32
2. Amendment No. 249 to DPR-37
3. Safety Evaluation

cc w/encls: See next page

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Package No. ML062920497
Amendment No. ML062920499
Tech Spec No. ML ML06300024

*Date of Safety Evaluation ** concurred date

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/AADB/BC	NRR/SCVB/BC	NRR/EEEB/BC	OGC	NRR/LPL2-1/BC
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DATE	10/11/06	10/12/06	8/25/2006	9/26/2006	10/3/2006	10/11/06	10/12/06

NRR/SSIB/BC
MScott**
10/10/2006

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 250
Renewed License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated January 31, 2006, as supplemented by letters dated February 23, June 21, and July 28, 2006, 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 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 250, are hereby incorporated in the renewed 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 by the end of the fall 2007 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: October 12, 2006

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated January 31, 2006, as supplemented by letters dated February 23, June 21, and July 28, 2006, 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 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated in the renewed 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 by the end of the fall 2006 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance: October 12, 2006

ATTACHMENT

TO LICENSE AMENDMENT NO. 250

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 249

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) 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
License	License
License No. DPR-32, page 3	License No. DPR-32, page 3
License No. DPR-37, page 3	License No. DPR-37, page 3
TSs	TSs
3.4-3	3.4-3
3.7-20	3.7-20
3.7-26	3.7-26
3.8-4	3.8-4
3.8-5	3.8-5
Figure 3.8-1	Figure 3.8-1
3.19-2	3.19-2
4.1-7	4.1-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 250 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 249 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated January 31, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060370098), as supplemented by letters dated February 23, (ADAMS Accession No. ML060540421), June 21, (ADAMS Accession No. ML061720499), and July 28, 2006 (ADAMS Accession No. ML062120719), Virginia Electric and Power Company (the licensee) submitted a request for changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), Technical Specifications (TSs). The licensee requested these TS changes as part of its resolution to the U.S. Nuclear Regulatory Commission (NRC) Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," and Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactor." Specifically, the licensee proposed to revise the method for starting the inside and outside recirculation spray (RS) pumps in response to a design-basis accident (DBA). Currently, the Surry 1 and 2 RS pumps start by using delay timers that are initiated when the containment pressure reaches the consequence limiting safeguards (CLS) High-High set point. The licensee's proposed change would result in starting the RS pumps by a coincident CLS High-High pressure and Refueling Water Storage Tank (RWST) level Low. In addition, the proposed changes would replace the current LOCTIC containment methodology with the Generation of Thermal-Hydraulic Information for Containments (GOTHIC) methodology described in Topical Report DOM-NAF-3, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment." These changes include revisions to the Loss-of-coolant accident (LOCA) alternate source term (AST) dose analysis that accommodates the changes to the RS pump start time.

The supplements dated February 23, June 21, and July 28, 2006, provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

NRC staff evaluated the proposed TS change against the dose criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR 50.67(b)(2)).

Implementation of an AST was previously reviewed and approved by the NRC staff in License Amendment Nos. 230 and 230 (ADAMS Accession No. ML020710159), dated March 8, 2002. These amendments addressed the impact of the proposed TS change on previously analyzed DBA radiological consequences and acceptability of the revised analysis results. The postulated LOCA is the only DBA affected by this proposed TS change, since only the LOCA analysis takes credit for operation of the RS.

Other relevant regulatory evaluations include the redefinition of an exclusion area boundary (EAB), which was reviewed and approved by the NRC staff in License Amendment Nos. 249 and 248 (ADAMS Accession No. ML062220194) on August 10, 2006, and the containment re-analysis presented in the NRC staff-approved Topical Report DOM-NAF-3 Revision 0, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment," (ADAMS Accession No. ML062420511) dated August 30, 2006.

The general design criteria (GDC) included in Appendix A to 10 CFR Part 50, did not become effective until May 21, 1971. The Construction Permits for Surry 1 and 2 were issued prior to May 21, 1971; consequently, these units were not subject to GDC requirements (Ref. SECY-92-223, dated September 18, 1992). The licensee stated that, however, the plant was designed to meet the intent of the draft GDC, including the following:

- *Criterion 38—Containment heat removal* which states that “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 that “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.”

The regulatory requirements upon which the NRC staff based its review, are the accident dose criteria in 10 CFR 50.67, and the guidance in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, and Sections 6.2.1, 6.2.1.1.A, 6.2.2 and 15.01 of NUREG-0800, “Standard Review Plan (SRP).” The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183.

3.0 TECHNICAL EVALUATION

Surry Power station (SPS) is a three-loop Westinghouse pressurized water reactor with a subatmospheric containment design. The engineered safeguards features that mitigate a LOCA or main steamline break accident (MSLB) event include the following (Chapters 5 and 6 of the SPS updated final safety analysis report (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, containment spray (CS) and RS, that operate together to reduce the containment temperature, return the containment pressure to subatmospheric, and remove heat from the containment. The RS subsystem maintains the containment subatmospheric and transfers heat from the containment to the service water (SW) system.

The CS system consists of two pumps that start on a CLS High-High containment pressure signal and draw suction from the RWST until the tank is empty. The RS system consists of four independent trains, each with one pump that takes suction from the containment sump. Two inside recirculation spray (IRS) pumps are located inside the containment sump, while two outside recirculation spray (ORS) pumps are located in the Safeguards Building. The RS pumps are started currently using delay timers that are initiated on the CLS High-High signal. Each RS train has a recirculation spray heat exchanger that is cooled by SW (on the tube side) for long-term containment heat removal. The SI system consists of two LHSI and three high-head safety injection pumps that draw from the RWST and inject into the reactor coolant system cold-legs. The SI pumps take suction from the RWST until a low-low level is reached, at which time recirculation mode transfer occurs. The recirculation mode transfer function changes the LHSI pump suction from the RWST to the containment sump, and the high-head safety injection pump suction from the RWST to the discharge header of the LHSI pumps.

Because the RS and SI systems use the containment sump to show that design criteria are satisfied, the resolution of NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (PWRs),” affects the operation of IRS, ORS and LHSI pumps. Appendix A to Regulatory Guide 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 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 net positive suction head (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 report, NEI-04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," (Ref. 1) also recommends the same criteria).

Currently, the SPS RS pumps start using delay timers that are initiated when the containment pressure reaches the CLS High-High setpoint. The IRS pumps have a 120-second setpoint and the ORS pumps have a 300-second setpoint. At these start times, the containment water level is predicted to be less than 1 ft in the current UFSAR containment analyses. While there is sufficient NPSH margin for the pumps, the current timer delay setpoints start the RS pumps when the sump strainer is partially submerged. Because the partial submergence requirement may be too restrictive for the sump strainer design, the licensee proposed delaying the RS pump start until sufficient water level is available in the containment.

The licensee proposed to start the IRS and ORS pumps on 60 percent RWST wide range (WR) level coincident with a CLS High-High containment pressure signal. The IRS pumps will receive an immediate start signal once the coincidence logic is satisfied. The ORS pumps will start using a 120-second delay timer from the coincident actuation signal. This delay will minimize the impact on emergency diesel generator loading and allow the IRS system to fill its piping completely, deliver spray to the containment, and reach a stable flow demand on the sump before the ORS pumps start. This method of starting the RS pumps ensures that a reliable mass of liquid has been added to the containment to meet the sump strainer submergence requirements for the range of LOCA break sizes that require the containment sump. The use of RWST WR level to start the RS pumps classifies the new instrumentation as part of the engineered safeguards features. Thus, the design will include safety-grade instrumentation consistent with UFSAR Section 7.5, "Engineered Safeguards" with surveillance requirements that must be added to the SPS TS.

The current SPS licensing basis analysis methodology for LOCA containment response is the Stone & Webster LOCTIC computer code that is described in SPS UFSAR Chapters 5 and 6. The licensee proposed to replace the LOCTIC methodology with GOTHIC analytical methodology that is described in topical report DOM-NAF-3, “GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment.” In a letter dated November 1, 2005, the licensee submitted DOM-NAF-3 to the NRC. On August 30, 2006, the NRC issued a safety evaluation on this topical report accepting the licensee’s GOTHIC containment analysis methodology.

The GOTHIC analyses in the licensee’s January 31 and July 28, 2006, submittals, are to replace the LOCTIC analyses in SPS UFSAR Chapters 5 and 6 for calculation of the following containment design requirements:

- LOCA peak containment pressure and temperature,
- LOCA containment depressurization time,
- LOCA containment peak pressure following depressurization,
- NPSH available for the LHSI pumps, and
- NPSH available for the ORS and IRS pumps.

Currently, the SPS UFSAR does not include LOCTIC containment response analyses for the MSLB event. The licensee performed MSLB containment response analyses with GOTHIC for introduction of explicit code calculations to the SPS licensing bases.
The licensee also used the minimum containment water level and maximum sump liquid temperatures from GOTHIC NPSH calculations to establish bounding inputs to the sump strainer design.

3.1 Radiological Consequences of the Affected Design-Basis Accident

The NRC staff performed independent confirmatory dose calculations for the LOCA using 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. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequences at selected receptors.

3.1.1 Loss-of-Coolant Accident

The licensee revised the LOCA analysis to reflect the effect of the proposed TS change on the current analysis, dated March 8, 2002 (ADAMS Accession No. ML020710159). The revised method for starting RS pumps will delay activation of the RS system, which will result in a short-term increase in air leakage from the containment and a short-term reduction in spray removal of radioactive fission products (FP) from the containment atmosphere.

To offset the potential increase in calculated radiological consequences, the licensee proposes the following changes to the LOCA analysis on record:

Proposed revised LOCA assumptions	Justification
Credit for filtration of the safeguards building exhaust by the auxiliary ventilation exhaust filter trains.	TS 3.22 and TS 4.12 assure 90 percent filter efficiency for elemental iodine, 70 percent for methyl (organic) iodine, and 99 percent for aerosol (particulate).
Credit for an increased CS flow rate in aerosol removal rate calculation.	Revised containment analysis based on NRC staff-approved Topical Report DOM-NAF-3.
Credit for RS iodine removal after CS termination.	Removal of current over-conservative assumption.
Reduction in containment volume sprayed after CS termination from 60 percent to 18 percent.	Arrangement of pumps, headers, and their emergency power supply guarantees RS only by 180 degrees of headers.
Emergency core cooling system (ECCS) leakage to start at 15 minutes vs the current 415 seconds.	Revised initiation logic such that outside RS pump will not start before 15 minutes.
ECCS leakage to auxiliary building and back leakage to RWST to start at 30 minutes versus the current 2300 seconds.	Revised initiation logic such that safety injection pumps will not start recirculating sump water before 30 minutes.
Containment leakage 1 to 4 hours after LOCA increases from 0.021 to 0.029 percent-volume/day.	Revised containment analysis based on NRC staff-approved Topical Report DOM-NAF-3.
Credit for revised EAB atmospheric dispersion factor of 1.79E-3 sec/m^3 versus the current 4.61E-3 sec/m^3.	See Section 3.1.3 below.

3.1.2 FP Removal by Containment Sprays

The CS system at Surry 1 and 2 consists of two separate parallel CS subsystems, each with 100 percent capacity, and four parallel RS subsystems, each with 50 percent capacity. The LOCA analysis takes credit for one CS subsystem and two RS subsystems.

The revised containment analysis, approved by the NRC staff on August 30, 2006, provided the timing of CS and RS operations, as well as the spray flow rates corresponding to the calculated rate of containment depressurization.

The licensee credited FP removal by the CS only until its termination at 1.14 hours. After that, the FP removal is credited by the RS only.

The licensee calculated the aerosol spray removal coefficients using a model described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," dated June 1993. This model is acceptable to the NRC staff, as noted in RG 1.183. The removal rate was calculated based on the 10th percentile level equation, which is considered by the NRC staff to be conservative, and therefore acceptable. The combined CS and RS revised removal coefficients are given in Table 1.

Table 1
Revised Combined CS and RS aerosol removal coefficients

Time (hr) from	Time (hr) to	lambda (hr^{-1})
2.78E-02	1.94E-01	3.59E+00
1.94E-01	5.56E-01	3.69E+00
5.56E-01	1.00E+00	4.16E+00
1.00E+00	1.14E+00	4.40E+00
1.14E+00	1.80E+00	2.53E+01
1.80E+00	1.84E+00	1.61E+01
1.84E+00	1.88E+00	1.12E+01
1.88E+00	2.07E+00	5.99E+00
2.07E+00	2.77E+00	3.25E+00
2.77E+00	3.57E+00	2.90E+00
3.57E+00	4.37E+00	2.86E+00
4.37E+00	7.20E+02	2.85E+00

The removal rate of elemental iodine by sprays was calculated using the same methodology as was previously approved by the NRC staff on March 8, 2002, and therefore, is acceptable.

3.1.3 Atmospheric Dispersion

The licensee used atmospheric dispersion factors (X/Q values) accepted by the NRC staff in previous license amendments. For the control room (CR) and low population zone (LPZ) dose estimates, the licensee used X/Q values reviewed and approved separately by the NRC staff on March 8, 2002, as part of the licensee's request to implement AST.

For the EAB dose estimate, the licensee used a X/Q value of 1.79×10^{-3} sec/m^3 that was approved by the NRC staff on August 10, 2006 (License Amendment Nos. 249 and 248). As part of the license amendment that revised the EAB, the EAB X/Q value was subsequently recalculated and approved as 1.76×10^{-3} sec/m^3. The licensee did not revise the EAB dose re-assessment because the use of 1.79x10 $^{-3}$ sec/m^3 results in estimation of a slightly higher dose than use of the actual EAB X/Q value of 1.76x10 $^{-3}$ sec/m^3.

Table 2
Atmospheric Dispersion Factors (X/Q values) for LOCA

EAB and LPZ

Exclusion Area Boundary (EAB)	0 - 2 hrs	1.79 x 10 $^{-3}$ sec/m^3 *
Low Population Zone (LPZ)	0 - 8 hrs	2.01 x 10 $^{-4}$ sec/m^3
	8 - 24 hrs	1.22 x 10 $^{-4}$ sec/m^3
	1 - 4 days	4.18 x 10 $^{-5}$ sec/m^3
	4 - 30 days	8.94 x 10 $^{-6}$ sec/m^3

* Note that the current Surry EAB X/Q value is 1.76 x 10 $^{-3}$ sec/m^3 as discussed above.

Control Room

Unit 1 Containment	0 - 2 hrs	6.74 x 10 $^{-4}$ sec/m^3
(to turbine building fresh air	2 - 8 hrs	5.18 x 10 $^{-4}$ sec/m^3
air intake # 1)	8 - 24 hrs	2.22 x 10 $^{-4}$ sec/m^3
	1 - 4 days	1.66 x 10 $^{-4}$ sec/m^3
	4 - 30 days	1.20 x 10 $^{-4}$ sec/m^3

Based on the previous license amendments dated March 8, 2002 (ADAMS Accession No. ML020710159), and August 10, 2006 (ADAMS Accession No. ML062220194), the NRC staff concludes that the CR, EAB and LPZ X/Q values presented in Table 2 are acceptable for use in the LOCA dose re-assessment .

3.1.4 Summary of Revised LOCA Analysis

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria provided in 10 CFR 50.67 and accident dose guidelines specified in SRP 15.0.1. The NRC staff 's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The assumptions found acceptable to the NRC staff are discussed in Section 3.1.1. The NRC staff finds that the EAB, LPZ, and CR doses that were estimated by the licensee for the LOCA, as presented in Table 3, meet the applicable dose criteria and are acceptable.

Table 3
Revised calculated DBA LOCA radiological doses

Total Effective Dose Equivalent

	CR	EAB	LPZ
Current dose consequences (rem)	4.86	24.01	3.57

Revised dose consequences	3.9	15.6	3.5
10 CFR 50.67 dose limits	5.0	25.0	25.0

3.1.5 TS Changes

In Surry TS 3.7, Table 3.7-2, under "7. RECIRCULATION MODE TRANSFER":

"a. RWST Level - Low" replaced by "a. RWST Level - Low-Low"

Also, added requirement "8. RECIRCULATION SPRAY":

"a. RWST Level - Low Coincident with High-High Containment Pressure"

“b. Automatic Actuation Logic and Actuation Relays”

In Surry TS 4.1-7, Table 4.1-1, under "15. RECIRCULATION MODE TRANSFER":

"a. RWST Level - Low" replaced by "a. RWST Level - Low-Low"

Also, added "16. RECIRCULATION SPRAY PUMP START":

"a. RWST Level - Low "

The proposed TS change is consistent with the Improved Standard Technical Specifications Change Traveler, TSTF-286-A, Revision 2, “Operations Involving Positive Reactivity Additions,” and NUREG-1431, “Westinghouse Owners Group Standard Technical Specifications,” Revision 3, dated March 31, 2004, and therefore, is acceptable.

3.1.6 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of a postulated LOCA with the proposed TS changes. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff also finds, with reasonable assurance, that the licensee’s estimates of the EAB, LPZ, and CR doses will comply with these criteria. Based on NRC staff’s independent evaluation, there is a reasonable assurance that Surry, as modified by this license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the NRC staff finds the proposed license amendment is acceptable with respect to the radiological consequences of LOCA.

3.2 Containment Analysis

The licensee proposed to revise its SPS containment analyses by converting from present Stone and Webster LOCTIC computer code to GOTHIC computer code. On August 30, 2006, the NRC issued a safety evaluation on this topical report accepting the licensee’s GOTHIC containment analysis methodology that is described in topical report DOM-NAF-3 (ADAMS Accession No. ML062420511).

The licensee used the GOTHIC code to perform containment analyses for the current RS system configuration with delay timers (120 seconds for IRS pumps, 300 seconds for ORS pumps) and the current TS Figure 3.8-1, containment air partial pressure limits, and compared the results with LOCTIC analyses. The licensee concluded that while some design inputs have changed from the LOCTIC analyses, transient behavior was similar to the LOCTIC UFSAR analyses.

The licensee used the GOTHIC code to perform containment analyses for the proposed configuration by making two changes in the above analysis. One change was to assume that the RS pumps started at 60 percent RWST water level coincident with a CLS High-High containment pressure signal. The IRS pumps were assumed to start directly from the signal while ORS pumps were assumed to start 120 seconds after receiving the actuation signal. The licensee included instrument uncertainty for the level signal and the timer setpoint. The other change was to increase the TS containment air partial pressure used in the analysis to the values in the proposed TS Figure 3.8-1, as shown in Figure 1 at the end of this report. These analyses were to show that adequate margins to the containment acceptance criteria from the following SPS UFSAR existed:

- LOCA and MSLB containment peak pressure < 45 psig
- LOCA containment pressure < 1.0 psig from 1 to 4 hours and < 0.0 psig after 4 hours
- LOCA containment temperature < 280 °F
- LHSI Pump NPSH available > NPSH required
- ORS Pump NPSH available > NPSH required
- IRS Pump NPSH available > NPSH required.

3.2.1 Application of the GOTHIC Methodology

Section 3.1 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the application of the GOTHIC methodology. The licensee stated that it used the containment response methodology described in topical report DOM-NAF-3 without modification for the GOTHIC analyses.

The licensee described modeling of the geometry, engineered safeguards features, containment passive heat sinks, plant parameter design inputs, containment initial conditions and instrument uncertainty, and NPSH available and water holdup. Table 3.1-1 of Attachment 1 to the licensee's letter, dated January 31, 2006, provided key parameters used in the containment analysis. The licensee used conservative assumptions consistent with DOM-NAF-3. For example, the licensee's assumptions in modeling NPSH available and water holdup for the LHSI, IRS and ORS pumps included the following:

- Used a multiplier of 1.2 to the direct diffusion layer model heat transfer coefficients for passive heat sinks. This lowers the prediction for the containment pressure by transferring more energy from the containment atmosphere to the passive heat sinks, and thus, conservatively lowers the predicted NPSH available.

- Assumed that all of the spray water is injected as droplets into the containment atmosphere (nozzle spray flow fraction of 1). This lowers the prediction for the containment pressure by transferring more energy from the containment atmosphere to the spray water, and thus, conservatively lowers the predicted NPSH available.

- Used the upper limit on containment free volume. This lowers the prediction of the containment pressure, and thus, conservatively lowers the predicted NPSH available.

- Used the minimum initial containment air pressure. This lowers the prediction of the containment pressure, and thus, conservatively lowers the predicted NPSH available.

- Used a minimum sump pool surface area for the containment volume liquid/vapor interface area. This minimizes the heat transfer from the sump water to the containment atmosphere because during recirculation sump pool water is hotter than the containment atmosphere. Therefore, this assumption results in predicting a higher sump water temperature and a lower containment pressure, and thus, a conservatively lower NPSH available.

In Section 3.1.4 of Attachment 1 to its letter, dated January 31, 2006, the licensee stated that it conservatively increased the surface area for metal heat sinks in containment based on a revised inventory that was documented in an internal calculation. The NRC staff requested the licensee to provide the old and new minimum surface area for metal heat sinks in containment because this information was not provided in the original submittal. In response, in a table in Attachment 2 to a letter, dated July 28, 2006, the licensee provided this information. The revised inventory has 23 percent higher surface area and 86 percent higher volume than the inventory used for the current analyses.

The NRC staff determined that the licensee's GOTHIC analyses are consistent with the NRC staff-approved topical report DOM-NAF-3 and is appropriate for analyzing the proposed changes.

3.2.2 Break Mass and Energy Releases

Section 3.2 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the modeling of the break mass and energy releases following a LOCA or MSLB. For the LOCA, the licensee applied the break release methodology in Section 3 of DOM-NAF-3. For the MSLB, the licensee conservatively used North Anna mass and energy release data from WCAP-11431, which were generated using the NRC-approved methodology from WCAP-8812 (Refs. 2 and 3, respectively). One change to the North Anna data was reducing the mass release rate from 900 gpm to 400 gpm after the faulted steam generator reached dryout to take credit for the SPS plant design which has cavitating venturis in the auxiliary feedwater (AFW) lines leading to each steam generator that limit the flow rate to about 350 gpm.

The NRC staff determined that the modeling of the break mass and energy releases were appropriate for GOTHIC containment analyses because the licensee used NRC-approved methodology.

3.2.3 LOCA Peak Pressure and Temperature

Section 3.3 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the calculation of the LOCA peak pressure and temperature. The LOCA peak containment temperature is related to the peak pressure because the containment atmosphere is saturated when the peak pressure occurs.

The double-ended hot-leg guillotine (DEHLG) break causes a more limiting blowdown peak pressure than the double-ended pump suction guillotine (DEPSG) break. GOTHIC code predicted the same peak temperature (273 °F) for DEHLG break for both current and proposed configurations. The GOTHIC code prediction for the peak pressure for the proposed configuration was higher than that for the current configuration (58.43 psia versus 57.17 psia for DEHLG break). This resulted from a higher TS containment air partial pressure assumed for the initial condition for the proposed configuration than the current configuration (11.3 versus 10.3 psia). The delay proposed for the starting of RS pumps would not affect the actual LOCA peak pressure and temperature which occur much earlier than the start of RS pumps.

For both current and proposed configurations, the containment peak pressure predicted is less than the containment design limit of 59.7 psia. The containment vapor temperature and liner temperature predicted were below the design limit of 280 °F.

Using acceptable conservative calculations, the licensee determined that the proposed maximum operating containment air partial pressure of 11.3 psia gives the LOCA peak pressure and temperature below the design limits. The NRC staff finds the licensee's LOCA peak pressure and temperature acceptable.

3.2.4 LOCA Containment Depressurization

Section 3.4 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the calculation of the LOCA containment depressurization. The depressurization analysis is performed to show that the containment can be returned to subatmospheric conditions consistent with the assumption for containment leakage in the dose consequences analysis. Currently, the UFSAR depressurization analyses using the LOCTIC code show that the containment is subatomospheric within 1 hour and remains subatmospheric thereafter.

The DEPSG break is limiting because it has the largest energy release to the containment due to the available energy removal from the steam generator secondary side. The licensee performed GOTHIC analyses to determine containment depressurization time (CDT) and the depressurization peak pressure (DPP) for a DEPSG break for both current and proposed configurations. For the current configuration, the containment pressure reached subatomospheric conditions at 2367 seconds (i.e., CDT) and the DPP was -1.38 psig at 5134 seconds. Therefore, the current configuration maintains a subatmospheric containment after 1 hour. For the proposed configuration, the licensee performed calculations for four different cases. Of these cases, Case 2 had limiting CDT of 3485 seconds and Case 4 had limiting DPP of 0.45 psig at 5121 seconds. These values are below the containment leak rate assumption values used for the dose analysis (Table 4.1-1 of Attachment 1 to the licensee's letter, dated January 31, 2006): pressure decreasing to 1.0 psig from 0 to 1 hour, 0.5 psig from 1 to 4 hours, and subatmospheric after 4 hours.

Using acceptable conservative calculations, the licensee determined LOCA containment depressurization parameters, CDT and DPP. Based on its review, NRC staff finds the licensee's calculation of CDT and DPP acceptable.

3.2.5 LHSI Pump NPSH Analysis

Section 3.5 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the LHSI pump NPSH analysis. The proposed delayed starting of RS pumps reduced the LHSI pump NPSH margin despite increasing the RS pump NPSH margin (see Section 3.2.6 of this report). The GOTHIC analysis predicted the LHSI Pump NPSH margin for the current and proposed configurations as 4.30 and 1.91 ft of water. Delaying the RS pump start reduces the system operating time before recirculation mode transfer from 2500 seconds to less than 1500 seconds. During this shorter window, lower SW temperature brings down the containment pressure quickly but the sump temperature holds up, lowering the NPSH available and thus, the NPSH margin.

Using acceptable conservative GOTHIC calculations, the licensee confirmed that the LHSI pump will perform its intended function. Based on its review, NRC staff finds the licensee's analysis acceptable.

3.2.6 RS Pump NPSH Analysis

Section 3.6 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the RS pump NPSH analysis which was performed using the GOTHIC code. The licensee stated that the ORS pump NPSH available is more limiting than the IRS pump NPSH available for three reasons: 1) IRS pump suction friction loss is 5.26 ft smaller (2.14 ft versus 7.4 ft for the ORS pump); 2) the ORS pump has 1.2 ft of extra head because the elevation of the pump impeller relative to the floor is -9.0 ft versus -7.8 ft for the IRS pump; and 3) the ORS pump suction receives 300 gpm of 45 °F RWST water, while the IRS pump receives 300 gpm of recirculation water from the heat exchanger discharge (hotter than the RWST). Items 2 and 3 provide more margin for the ORS pump, but this amount is more than offset by the smaller suction friction loss for the IRS pump (Item 1). The licensee confirmed that ORS pump NPSH available was limiting by running a case to minimize IRS pump NPSH available (maximum IRS pump flow and minimum ORS pump flow).

For the current configuration, the licensee analyzed the DEHLG break, which is limiting because the mass and energy data maximize the energy in the containment sump early in the accident. For the proposed configuration with delayed RS pump start, the licensee analyzed both DEHLG and DEPSG breaks for a range of single failures.

For the current configuration, the licensee calculated IRS and ORS pump NPSH available of 14.03 ft and 10.68 ft of water, giving corresponding NPSH margins of 3.53 ft and 1.49 ft of water. For the proposed configuration, the licensee calculated a minimum IRS and ORS pump NPSH available of 15.54 ft and 12.88 ft of water, giving corresponding NPSH margins of 5.04 and 3.69 ft of water.

Using acceptable conservative calculations, the licensee confirmed that the RS pump will perform its intended function. Based on its review, NRC staff finds the licensee's analysis acceptable.

3.2.7 MSLB Peak Pressure and Temperature

Section 3.7 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the MSLB peak pressure and temperature analyses. SPS UFSAR Section 6.2.2.1.4, "Steam-Line Break Protection," refers to MSLB analysis performed for Beaver Valley which was bounding for SPS. The licensee performed MSLB peak pressure and temperature analyses for SPS to remove reference to the Beaver Valley analysis.

The SPS GOTHIC analyses for MSLB do not credit the recirculation spray system. This precludes the need to show adequate RS pump performance during an MSLB event. The limiting single failure in the containment model is the loss of an emergency bus, leaving one CS pump available with minimum flow and maximum time to deliver spray to containment.

For the current and proposed configurations, the GOTHIC MSLB analyses predicted containment peak pressure of 58.12 and 59.48 psia. These values are more limiting than those predicted in the LOCA analyses described in Section 3.2.3 of this report. The main conservatism is the assumption for AFW flow rate in the North Anna mass and energy release analysis used for the SPS analyses as described in Section 3.2.2 of this report. The North Anna analysis assumed 900 gpm AFW flow rate to the faulted steam generator and 350 gpm to each of the two intact steam generators. SPS has cavitating venturis in the AFW lines leading to each SG that limit the flow rate to about 350 gpm. The difference in mass flowrate between the physical limit of 350 gpm and the assumed value of 900 gpm delays the time for the faulted SG to reach dryout, more energy is released to the containment, resulting in a higher peak pressure. SPS-specific lower AFW flow rates would lead to a smaller mass release from the break and a lower peak pressure. For the MSLB analyses, the licensee reduced the mass release rate from 900 to 400 gpm after the faulted SG reached dryout. This assumption provided a conservative, but reasonable long-term containment pressure and temperature response for SPS but did not affect the containment peak pressure and temperature, which occur earlier in the event.

For current and proposed configurations, the licensee calculated the MSLB peak containment air temperature of 324.5 and 318.9 °F for current and proposed configurations, respectively, which are higher than the design limit of 280 °F. The licensee's analyses included an additional 1 ft^2 thermal conductor to determine a conservative containment liner temperature response in accordance with Section 3.3.3 of DOM-NAF-3. The conductor used a 1.2 multiplier on the direct diffusion layer model heat transfer coefficient. The peak liner temperature for the proposed configuration was 251.1 °F at 490 seconds, and therefore, the MSLB peak containment air temperature exceeding the containment design limit did not adversely affect the containment liner. Section 3.2.9 of this report describes the effect and concludes that the MSLB peak containment air temperature exceeding the design limit will not adversely affect the safety-related equipment inside the containment.

Using acceptable conservative calculations, the licensee calculated (1) MSLB peak containment pressure, which is bounded by the containment design peak pressure and (2) MSLB peak containment air temperature, which exceeded the containment design temperature. The licensee showed that the MSLB peak containment air temperature exceeding design limit did not adversely affect the containment liner or the safety-related equipment inside the containment. Based on its review, NRC staff finds the licensee's analysis acceptable.

3.2.8 Inadvertent CS Actuation Event

Section 3.8 of Attachment 1 to the licensee's letter, dated January 31, 2006, describes the inadvertent CS actuation analysis. SPS UFSAR Section 5.3.4.3 describes the containment response using the CONTEMPT4/MOD5 code for an inadvertent CS actuation analysis. The licensee proposed to increase the TS containment air partial pressure minimum limit from 9.0 to 10.1 psia, as described in Section 3.8 of Attachment 1 to the licensee's letter, dated January 31, 2006, and to eliminate the CONTEMPT4/MOD5 containment response analysis and the time-critical operator action for the inadvertent CS actuation event. The licensee is to replace the UFSAR analysis with an application of the equation of state for an ideal gas (Charles' Law). This methodology is identical to the application in North Anna UFSAR, Section 6.2.6.3. The licensee's analysis show that at the TS minimum air partial pressure (10.1 psia) and without crediting an operator action, the containment mat liner pressure did not reach the SPS TS 5.2 limit of 8.0 psia.

The NRC staff reviewed and verified the licensee's calculation of the final containment pressure following an inadvertent CS actuation analysis for replacing the CONTEMPT4/MOD5 containment analysis given in SPS UFSAR and provides basis for eliminating the time-critical operator action following such actuation.

3.2.9 Equipment Qualification (EQ)/EQ Envelope Verification

In its submittal dated January 31, 2006, the licensee stated that it had developed new pressure and temperature EQ envelopes that sufficiently bounded the GOTHIC LOCA and the MSLB pressure and temperature profiles. By letter dated July 28, 2006, the licensee indicated that the composite pressure and temperature profiles were developed from the LOCA and MSLB pressure and temperature profiles from the NRC staff-approved GOTHIC methodology. These composite profiles were then compared to the EQ test reports for all environmentally qualified equipment inside the containment. As a result of this comparison, the licensee concluded that the environmentally qualified equipment inside the containment was qualified for the GOTHIC accident analysis profiles for pressure and temperature.

Therefore, based on the licensee's assessment documented in the submittal on January 31, 2006, NRC staff concluded that the GOTHIC accident profile is enveloped by the EQ test report. The EQ status of equipment inside containment is not affected by the revised containment temperature and pressure profiles resulting from delaying RS pump start and increasing the containment air partial pressure limits. The NRC staff finds that the licensee's EQ envelope verification is acceptable and complies with 10 CFR 50.49.

3.2.10 Proposed TS Limits for Containment Air Partial Pressure versus SW Temperature

In Section 3.10 of Attachment 1 to its letter, dated January 31, 2006, the licensee describes the proposed TS limits for containment air partial pressure versus SW temperature as given in TS Figure 3.8-1, which is shown in Figure 1.

This operating domain maintains the current limits of 25-95 °F for service water temperature and 75-125 °F for containment temperature, and accounts for 0.25 psi instrument uncertainty for the containment air partial pressure. The following defined the allowable containment air partial pressure limits as shown on Figure 1:

- The MSLB peak pressure analysis limits the maximum operating air partial pressure to 11.3 psia. The LOCA peak pressure (58.43 psia for the proposed configuration, Section 3.2.3) is less than the MSLB peak pressure (59.48 psia for the proposed configuration, Section 3.2.7). The MSLB analysis sets the TS upper limit between 25 and 70 °F SW temperature.

- The containment depressurization analyses (Section 3.2.4) set the TS upper limit from 11.3 psia at 70 °F service water to 10.3 psia at 95 °F service water. The allowable air pressure decreases with increasing service water temperature because it is more difficult to depressurize the containment at higher service water temperature. To meet subatmospheric requirements, the initial air partial pressure is limited to 10.3 psia at 95 °F SW.

- The LHSI pump NPSH analyses (Section 3.2.5) set the lower limit on air partial pressure (the RS pumps use the same assumptions but have more NPSH margin (Section 3.2.6)). At the same air partial pressure, the NPSH analyses are limiting at 25 °F SW. Therefore, the TS lower limit is sloped below 70 °F to recover LHSI pump NPSH margin. The proposed lower limit in Figure 1 ensures at least 1.6 ft of NPSH margin across the entire SW temperature range.

The NRC staff determined that the licensee's analyses described in Sections 3.2.3 through 3.2.7 of this report provide bases for the proposed changes to TS Figure 3.8-1.

3.2.11 Summary of Containment Analysis Results

In Table 3.11-1 of Attachment 1 to its letter, dated January 31, 2006, which is given as Table 4 in this report, the licensee summarized the containment analysis results and compared them to the design limits.

Table 4
GOTHIC Containment Analysis Results

Acceptance Criterion	Design Limit	Current Configuration	Proposed Configuration
LOCA Peak Pressure	59.7 psia	57.17 psia	58.43 psia
LOCA Peak Temperature	280 °F	273.0 °F	273.0 °F
MSLB Peak Pressure	59.7 psia	58.12 psia	59.48 psia
MSLB Peak Temperature*	280 °F	324.5 °F	318.9 °F
Containment Depressurization Time	< 1.0 psig at 1 hour	2357 sec to < 0.0 psig	3485 sec to < 0.0 psig

Depressurization Peak Pressure	< 1 .0 psig 1-4 hours	-1.38 psig	+0.45 psig
LHSI Pump NPSH Available	13.82 ft at 3330 gpm	18.12 ft	15.73 ft
ORS Pump NPSH Available	9.19 ft at 3300 gpm	10.68 ft	12.88 ft
IRS Pump NPSH Available	10.5 ft at 3650 gpm	14.03 ft	15.54 ft

* See Section 3.2.9 of this report for the disposition of MSLB peak temperature exceeding the design limit.

The results of the GOTHIC analyses for the proposed configuration show that all containment analysis acceptance criteria, except for the MSLB peak temperature (see Section 3.2.9 of this report for the disposition of MSLB peak temperature exceeding the design limit), are met for operation in the allowable region of Figure 1 and starting the RS pumps on 60 percent RWST WR level coincident with CLS High-High containment pressure. GOTHIC MSLB temperatures greater than 280 °F does not adversely affect the integrity of the containment liner or the operation of safety-related equipment inside containment as discussed above.

The licensee proposed operation in the allowable region of Figure 1 and starting the RS pumps on 60 percent RWST WR level coincident with CLS High-High containment pressure. The licensee showed using the NRC-approved "GOTHIC Methodology for Analyzing the Response to Postulated Ruptures Inside Containment," that the proposed changes would not affect the containment analyses acceptance criteria (NRC staff SE, dated August 30, 2006). The NRC staff determined that the proposed changes meet (1) GDC 38 because the licensee showed that the containment sprays would remove containment heat to reduce containment pressure and temperature rapidly, consistent with the functioning of other associated systems, following any LOCA and maintain them at acceptably low levels and (2) GDC 50 because the licensee showed that the reactor containment structure, including access openings, penetrations, and the containment heat removal system can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. Therefore, the proposed license amendment is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 13182, March 14, 2006). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

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

2. WCAP-11431, Revision 0, “Mass and Energy Releases Following a Steam Line Rupture for North Anna Units 1 and 2,” February 1987.

3. WCAP-8822-P, “Mass and Energy Releases Following a Steam Line Rupture,” September 1976, with Supplements 1 (WCAP-8822-S1-P-A) and 2 (WCAP-8822-S2-P-A) both dated September 1986 (WCAP-8860 is the Non-Proprietary version).

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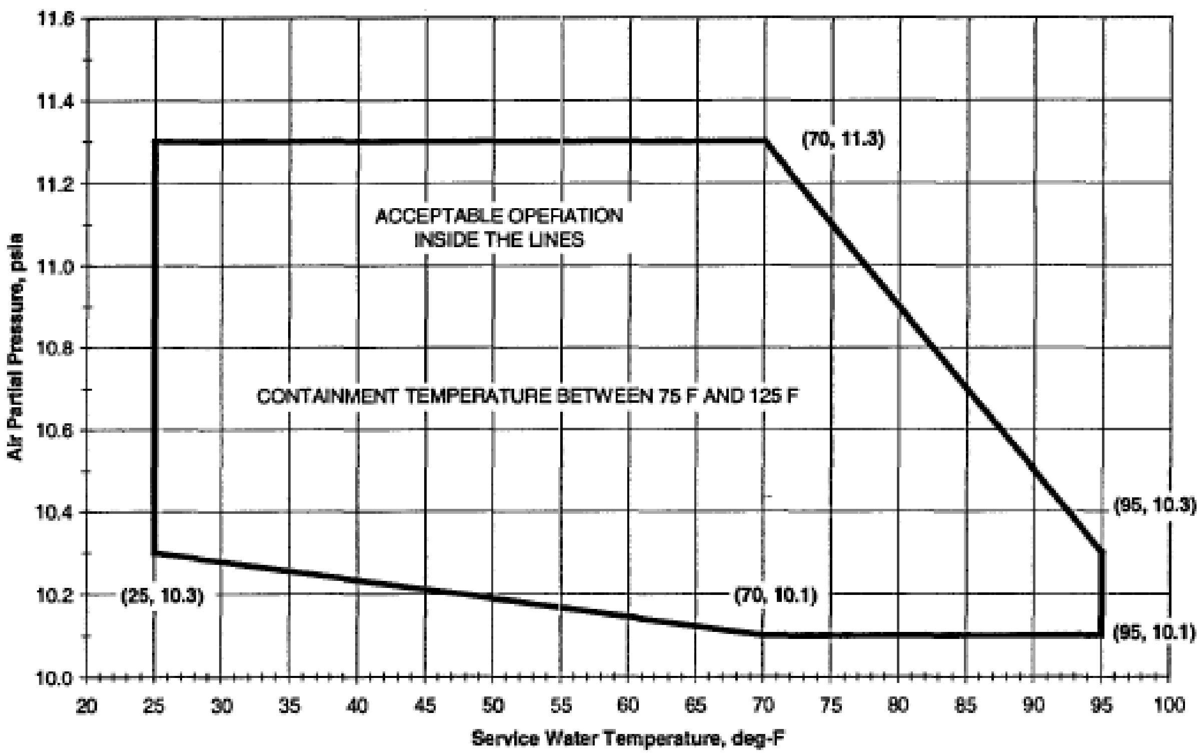

Figure 1. Containment air partial pressure versus service water temperature (proposed TS Figure 3.8-1)

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