VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

January 27, 2010

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	09-223
NLOS/GDM	R1
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNITS 1 AND 2
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

Pursuant to 10 CFR 50.90, Dominion requests amendments to Operating Licenses DPR-32 and DPR-37 for Surry Power Station Units 1 and 2 respectively. This measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) would increase each unit's rated power (RP) level from 2546 megawatts thermal (MWt) to 2587 MWt, and make Technical Specifications changes as necessary to support operation at the uprated power level. The proposed change is an increase in RP of approximately 1.6%.

Dominion developed this LAR utilizing the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." The proposed uprate is characterized as a MUR using the Cameron (formerly Caldon) Leading Edge Flow Meter (LEFM) CheckPlus System to improve plant calorimetric heat balance measurement accuracy. In addition, NRC requests for additional information (RAIs) associated with MUR applications for other nuclear units were reviewed for applicability. Information addressing many of those RAIs is included in Attachment 5.

The proposed changes have been reviewed and approved by the Facility Safety Review Committee.

Information provided in the attachments to this letter is summarized below:

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NRR

- Attachment 1 provides Description, Technical Analysis, Regulatory Analysis and Environmental Analysis for the proposed Operating License and Technical Specifications changes. As discussed in this attachment, the proposed amendment does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92. The Facility Safety Review Committee has reviewed and concurred with this determination.

- Attachment 2 contains marked-up pages to reflect the proposed changes to the Operating Licenses and Technical Specifications.

- Attachment 3 contains typed pages to reflect the proposed changes to the Operating Licenses and Technical Specifications.

- Attachment 4 contains marked-up pages to reflect the proposed changes to the Technical Specifications Bases and Technical Requirements Manual. These changes are provided for information only.

- Attachment 5 provides the information recommended for inclusion in a MUR LAR submittal by NRC RIS 2002-03. This information demonstrates acceptable plant operation at the increased RP of 2587 MWt.

- Attachment 6 lists the regulatory commitments associated with this LAR.

Additional information required to support the license amendment request that has been determined to be proprietary in accordance with 10 CFR 2.390 is being submitted under separate cover letter (Serial No. 09-223A dated January 27, 2010). This letter includes the proprietary Cameron Bounding Uncertainty Analysis Reports and the Cameron Meter Factor Reports for Surry Units 1 and 2, as well as the supporting request for withholding from public disclosure and associated affidavit. In addition, as discussed with the NRC staff during a January 12, 2010 conference call, a second supporting submittal is also being provided to address safety analysis updates that were identified during the Surry MUR technical review process and were determined to require NRC review pursuant to 10 CFR 50.90 (Serial No. 09-223B).

Dominion requests approval of the proposed amendments by August 31, 2010 with a 90-day implementation period. This will accommodate Dominion's plans to implement the MUR by November 30, 2010.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request, with attachments, is being provided to the designated State of Virginia official.

If you have any questions or require additional information, please contact Mr. Gary Miller at (804) 273-2771.

Sincerely,

L. N. Hartz
Vice President - Nuclear Support Services

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President – Nuclear Support Services, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 27TH day of January, 2010.

My Commission Expires: May 31, 2010.

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2010

Vicki L. Hull
Notary Public

Attachments:

1. Discussion of Change
2. Proposed Operating License and Technical Specifications Pages (Marked-up)
3. Proposed Operating License and Technical Specifications Pages (Typed)
4. Proposed Technical Specifications Bases and Technical Requirements Manual Pages (Marked-up) For Information Only
5. NRC Regulatory Issue Summary 2002-03 Requested Information
6. List of Regulatory Commitments

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

DISCUSSION OF CHANGE

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

1.0 DESCRIPTION

Virginia Electric and Power Company (Dominion) proposes a change to the Surry Power Station (SPS) Units 1 and 2 Operating Licenses pursuant to 10 CFR 50.90. The measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) would increase each unit's rated power (RP) from 2546 megawatts thermal (MWt) to 2587 MWt and make Technical Specification changes as necessary to support operation at the uprated power level. The proposed change is an increase in RP of approximately 1.6%. Unless otherwise noted, 100% power in this LAR refers to 2587 MWt.

This LAR is based on installing and utilizing the Cameron (formally known as Caldon) Leading Edge Flow Meter (LEFM) CheckPlus System as an ultrasonic flow meter (UFM) located in each of the three main feedwater lines supplying the steam generators. The Dominion nomenclature for the Cameron LEFM CheckPlus System is often simplified to feedwater ultrasonic flow meter or UFM.

The original means of measuring feedwater flow using venturis will remain in place performing their original instrumentation protection and control functions. The UFM will be used as the primary method of determining the feedwater flow rate in the plant's calorimetric heat balance and the venturi based feed or steam flow rate will become the backup method. The justification for an increase in licensed RP is based on the increased accuracy of the UFM.

Dominion evaluated the impact of the SPS MUR uprate to 2587 MWt on applicable systems, structures, components, and safety analyses. Dominion determined that no significant hazards consideration exists as defined by 10 CFR 50.92. In addition, Dominion concluded that the proposed change qualifies for categorical exclusion from performing an environmental assessment as set forth in 10 CFR 51.22(c)(9); therefore, no environmental impact statement or environmental assessment is included or needed for approval.

2.0 BACKGROUND

SPS was initially licensed to operate at a maximum of 2441 MWt. In Amendment 203, dated August 3, 1995, the Nuclear Regulatory Commission (NRC) approved SPS operation at the current power level of 2546 MWt. The proposed MUR power uprate is based on a redistribution of analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed per the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Appendix K originally mandated 102% of licensed power level for light water reactor ECCS evaluation models. The NRC approved a change to the 10 CFR 50, Appendix K requirements on June 1, 2000 effective July 31, 2000. This change provided licensees the option of maintaining the 2% power margin between licensed power level and the ECCS evaluation assumed power level, or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

Implementing the feedwater UFM (Cameron LEFM CheckPlus System) is an effective way to obtain additional plant power without significantly affecting plant design margins. Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The UFM provides a more accurate measurement of feedwater flow and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.35% at RP.

The UFM will provide on-line main feedwater flow and temperature measurement to determine reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The UFM consists of a measuring section containing 16 ultrasonic multi-path transit time transducers, one dual resistance temperature detector (RTD), and two pressure transmitters installed in each of the three feedwater lines, and an electronic signal processing cabinet.

The UFM will be used in lieu of the current steam or feed flow venturi indication and RTD temperature indication to perform the plant calorimetric measurement calculation. The currently installed steam or feed flow instruments will continue to provide inputs to other indication, protection and control systems, and will be used if the UFM is not functional.

3.0 PROPOSED CHANGE

The proposed (marked-up) Operating License (OL) and Technical Specifications (TS) changes are provided in Attachment 2. The typed OL and TS pages are provided in Attachment 3.

The proposed OL and TS changes are described below.

Renewed Facility Operating License - Maximum Power Level

Paragraph 3.A, "Maximum Power Level," of the Unit 1 and Unit 2 Operating Licenses (DPR-32 and DPR-37 respectively) authorizes facility operation at a reactor core power level not in excess of 2546 megawatts (thermal). The proposed change increases the Maximum Power Level from its current value of 2546 MWt to 2587 MWt.

TS Section 1.0, Definitions – Rated Power

The Technical Specification definition of RATED POWER (RP) limits the reactor core power level to 2546 MWt. The MUR power uprate is equivalent to an approximately 1.6% increase in the current RP. The RP definition is revised to change the value from 2546 MWt to 2587 MWt, for consistency with the Maximum Power Level in Renewed Facility Operating License Paragraph 3.A.

TS Figure 2.1-1, Reactor Core Thermal and Hydraulic Safety Limits

Changes are required to the Reactor Core Thermal and Hydraulic Safety Limits in TS Figure 2.1-1. Revised Reactor Core Safety Limits (RCSLs) have been developed to reflect MUR operating conditions. The RCSLs are defined as the most limiting of vessel exit boiling, hot channel exit quality, and the core Departure from Nucleate Boiling considerations. The RCSLs were evaluated using the approved Westinghouse methodology described in WCAP-8745-P-A, *Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,* September 1986.

TS Section 2.3.A.2(d) - Overtemperature ΔT

A change to the Overtemperature ΔT (OTΔT) trip pressure constant, K3, is required to ensure protection at low Reactor Coolant System (RCS) pressures (i.e., down to the low pressure reactor trip Technical Specification 2.3.A.2(c) setting limit of 1875 psig) for MUR conditions. The new setpoint equation was determined in accordance with the methodology of WCAP-8745-P-A.

Technical Requirements Manual

Technical Requirements Manual (TRM) changes are being made to support this LAR. These changes consist of the addition of a new TRM section for UFM. The TRM changes are provided for information only in Attachment 4. The TRM requirements are described in Updated Final Safety Analysis Report (UFSAR) Section 16.2. As stated in UFSAR Section 16.2, TRM changes are controlled using the 10 CFR 50.59 process.

Updated Final Safety Analysis Report

Changes to the SPS UFSAR are being made to support this LAR. These changes will be made in accordance with 10 CFR 50.59.

4.0 TECHNICAL ANALYSIS

SPS Units 1 and 2 are presently licensed for an RP of 2546 MWt. Using more accurate feedwater flow measurement equipment supports an approximately 1.6% increase to 2587 MWt. The power uprate evaluations addressed the following categories: nuclear steam supply system (NSSS) performance parameters, accidents, design transients, systems, components, nuclear fuel, and interfaces between NSSS and balance-of-plant (BOP) systems. The evaluation conclusions are summarized in Attachment 5, Requested Information NRC Regulatory Issue Summary (RIS) 2002-03. These analyses were reviewed to provide assurances that they remain bounding for the proposed power uprate. Non-bounding analyses are discussed in Attachment 5, Section III.

Table 4.0-1 indicates the power levels used for the SPS MUR power uprate analyses and evaluations. Each area of analysis scope assumed an appropriate core power that bounds the proposed 2587 MWt value (nominal or nominal plus uncertainty).

Table 4.0-1
Analysis Power Levels for Surry Units 1 and 2 MUR Uprating

Analysis Scope	Core Power MWt	NSSS Power MWt(3)	Source
NSSS	2597(1)	2609	NSSS Design Parameters
Safety Analyses	2596.9(1)	2609	UFSAR Chapters 5, 6 and 14
Statistical DNBR Events	2589.3(2)	2602	UFSAR Chapter 14
Safety-Related System Evaluations	2597(1)	2609	Consistent with UFSAR safety analyses
BOP System Evaluations	2589.3(2)	2602	

1. 102% of current RP of 2546 MWt; while many safety analyses assume 2597 MWt, some safety analyses assume 2596.9 MWt, which is identified as the limiting analysis value in the table.
2. 101.7% of current RP of 2546 MWt
3. The analyses use 12 MWt for RCP net heat addition. For the BOP system evaluations, the NSSS power is rounded up to the next whole number

The NSSS design thermal and hydraulic parameters derived from the power uprate conditions serve as the basis for the NSSS analyses. A detailed review of the accident analyses was performed for the steam generator tube rupture, loss of coolant accident (LOCA), and non-LOCA areas. The currently assumed loss of coolant mass and energy release remains bounding. The fuel was evaluated for its ability to perform at the uprated power level. Dominion concludes that the changes to the SPS design basis and transient analyses are acceptable. Each of the NSSS systems and components was evaluated at the uprated conditions. The BOP systems, electrical power systems, control systems and instrumentation systems were also evaluated at the uprated conditions. The analyses and evaluations performed demonstrate that the acceptance criteria continue to be met. SPS Units 1 and 2 require minimal plant modifications to safely operate at the uprated conditions (Attachment 5 Section VII.2.B).

4.1 Nuclear Steam Supply System Design Parameters

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. The design parameters are established using conservative input assumptions to provide bounding conditions used in the NSSS analyses. They provide the primary and secondary side system conditions (thermal power, temperatures, pressures, flow) that are used as the basis for the NSSS analyses and evaluations. These parameters were revised to account for the increase in analyzed core power from 2546 MWt to 2597 MWt. The new parameters are listed in Table 4.0-2. These parameters have been incorporated, as required, into the applicable NSSS system and component evaluations and safety analyses performed to support the power uprate.

4.2 Input Parameters

The major input parameters used to calculate the four cases of NSSS design parameters are as follows:

- NSSS power level of 2609 MWt (2597 MWt core power plus 12 MWt reactor coolant pump (RCP) net heat input).
- Core bypass flow of 6.0%, which accounts for thimble plug removal.
- Feedwater temperature of 452.0°F.
- Westinghouse Model 51F replacement steam generators (SG).
- Vessel average temperature (T_{avg}) range of 570.0°F to 576.0°F, which envelopes the current T_{avg} of 573.0°F.

- Maximum steam generator moisture carryover of 0.25%.
- Steam generator tube plugging (SGTP) levels of 0% and 7%.
- Thermal design flow maintained at 88,500 gpm/loop.
- Reactor coolant pressure of 2250 psia, which is the current operating value.

4.3 Parameter Cases

Four cases of NSSS design parameters were used to evaluate the power uprate impact. These four cases are shown in Table 4.0-2.

Cases 1 and 2 represent parameters applicable to most NSSS analyses that are based on the minimum T_{avg} of 570.0°F. Case 2 is based on an average 7% SGTP and yields the minimum SG secondary side steam pressure and temperature. Note that the primary side temperatures are identical for these two cases.

Cases 3 and 4 represent parameters applicable to most NSSS analyses that are based on the maximum T_{avg} of 576.0°F. Case 3 is based on an average 0% SGTP and yields the maximum SG secondary side steam pressure and temperature. Note that the primary side temperatures are identical for these two cases.

The various NSSS analyses and evaluations (e.g., systems, components and materials) performed for the MUR power uprate incorporated the design parameters appropriate for those analytical areas.

Table 4.0-2
NSSS Design Parameters for Surry Units 1 and 2 MUR Uprating

		Bounding 2% Uprate			
Parameter	**Current**	**Case 1**	**Case 2**	**Case 3**	**Case 4**
THERMAL DESIGN					
NSSS Power, MWt	2558(2)	2609	2609	2609	2609
10^6 BTU/hr	8715	8902	8902	8902	8902

Table 4.0-2 (Continued)
NSSS Design Parameters for Surry Units 1 and 2 MUR Uprating

Parameter	Current	Bounding 2% Uprate Case 1	Case 2	Case 3	Case 4
Reactor Power, MWt	2546	2597	2597	2597	2597
10^6 BTU/hr	8687	8861	8861	8861	8861
Thermal Design Flow, Loop gpm	88,500	88,500	88,500	88,500	88,500
Reactor 10^6 lbm/hr	101.1	101.6	101.6	100.8	100.8
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.0(1)	6.0(1)	6.0(1)	6.0(1)	6.0(1)
Reactor Coolant Temperature, °F					
Core Outlet	609.3	607.2	607.2	612.8	612.8
Vessel Outlet	605.6	603.3	603.3	609.1	609.1
Core Average	576.5	573.6	573.6	579.6	579.6
Vessel Average(3)	573.0	570.0	570.0	576.0	576.0
Vessel/Core Inlet	540.4	536.7	536.7	542.9	542.9
Steam Generator Outlet	540.1	536.3	536.3	542.6	542.6
Steam Generator					
Steam Temperature, °F	515.9	511.3	508.0	517.9	515.4
Steam Pressure, psia	784	753	737	798	781
Steam Flow, 10^6 lbm/hr total	11.26	11.62	11.61	11.64	11.63
Feedwater Temperature, °F	443	452	452	452	452
Moisture, % maximum	0.25	0.25	0.25	0.25	0.25
Steam Generator Tube Plugging, %	7	0	7	0	7
Zero Load Temperature, °F	547	547	547	547	547
HYDRAULIC DESIGN					
Mechanical Design Flow, gpm	100,300	100,300	100,300	100,300	100,300

1. Core bypass flow includes 1.5% due to thimble plug removal.
2. This represents the current NSSS analyzed power level of 2546 MWt core power plus 12 MWt for RCP net heat input.
3. The MUR uprate program analyzed a full-power T_{avg} window of 570–576°F. Surry plans to continue operation at a full-power T_{avg} of 573°F.

5.0 REGULATORY ANALYSIS

Dominion has evaluated the License Amendment Request (LAR) against the 10 CFR 50.92 criteria to determine if any significant hazards consideration is involved. Dominion has concluded that this LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

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

   Response: No.

   The proposed change will increase the Surry Power Station (SPS) Units 1 and 2 rated power (RP) from 2546 megawatts thermal (MWt) to 2587 MWt. Nuclear steam supply system and balance-of-plant systems, components and analyses that could be affected by the proposed change to the RP were evaluated using revised design parameters. The evaluations determined that these structures, systems and components are capable of performing their design function at the proposed uprated RP of 2587 MWt. An evaluation of the accident analyses demonstrates that the applicable analysis acceptance criteria are still met with the proposed changes. Power level is an input assumption to equipment design and accident analyses, but it is not a transient or accident initiator. Accident initiators are not affected by the power uprate, and plant safety barrier challenges are not created by the proposed changes.

   The radiological consequences of operation at the uprated power conditions have been assessed. The proposed change to RP does not affect release paths, frequency of release, or the analyzed reactor core fission product inventory for any accidents previously evaluated in the SPS Updated Final Safety Analysis Report. There is a small increase in the reactor coolant activity concentration. Structures, systems and components required to mitigate transients are capable of performing their design functions with the proposed changes, and are thus acceptable. Analyses performed to assess the effects of mass and energy releases remain valid. The assessment of radiological consequences for operation at the proposed power level confirmed that there is not a significant increase for affected events.

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

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

   Response: No.

   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any proposed changes. The ultrasonic flow meter (UFM) being installed to facilitate the Measurement Uncertainty Recapture (MUR) power uprate has been analyzed, and system failures will not adversely affect any safety-related system or any structures, systems or components required for transient mitigation. Structures, systems and components previously required for transient mitigation are still capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

   The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at an RP of 2587 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

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

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

   Response: No.

   The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Core analyses demonstrate that power uprate implementation will continue to meet the current nuclear design basis. Impacts to components associated with the reactor coolant system pressure boundary structural integrity, and factors such as pressure-temperature limits, vessel fluence, and pressurized thermal shock were determined to be bounded by the current analyses.

   Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions following implementation of the proposed change. The current SPS safety analyses, and the revised design basis radiological accident dose calculations, bound the power uprate without significantly impacting margins.

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

Based on the above, Dominion concludes that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and a finding of no significant hazards consideration is acceptable.

6.0 ENVIRONMENTAL ANALYSIS

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

Dominion has concluded that this license amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment is required in connection with issuance of the proposed license amendment. This determination is based on the following:

(i) The license amendment request does not involve a significant hazards consideration, as described in the significant hazards evaluation.

(ii) The proposed change does not involve installing new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite.

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level.

Gaseous and liquid radwaste effluent activity is expected to increase from current levels by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and10 CFR 50, Appendix I limits per the SPS Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

Solid radwaste effluent activity is expected to increase from current levels proportionately to the increase in long half-life coolant activity. The total long-lived activity is bounded by the percent of power uprate. Changes in solid waste volume are not expected.

Therefore, the proposed license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

(iii) The license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage of core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within acceptable limits by the site Radiation Protection Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

7.0 PRECEDENT

License amendment applications based on the Cameron (formerly Caldon) LEFM CheckPlus system were previously approved for PWRs North Anna 1 and 2 (Reference 8.1) Seabrook Station (Reference 8.2), Crystal River 3 (Reference 8.3) and Vogtle 1 and 2 (Reference 8.4). These submittals requested NRC approval to increase licensed power level by reducing uncertainty through the use of the LEFM CheckPlus system for feedwater flow measurement. The Surry Units 1 and 2 submittal is comparable to those license amendment requests.

8.0 REFERENCES

8.1 NRC letter to Virginia Electric and Power Company, *North Anna Power Station, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME0965 and ME0966)*, ML092250616, October 22, 2009.

8.2 NRC letter to FPL Energy Seabrook, *Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML061360034, May 22, 2006.

8.3 NRC letter to Crystal River 3, *Crystal River 3 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML073600419, December 26, 2007.

8.4 NRC letter to Vogtle Electric Generating Plant, Units 1 and 2, *Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate*, ML080350347, February 27, 2008.

ATTACHMENT 2

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES (MARKED-UP)

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).

      2587

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 265 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications. (1)

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2546 megawatts (thermal).
   2587

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 264, are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications. 1

   C. Reports

   The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

   The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 54

   F. Deleted by Amendment 59 and Amendment 65

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. RATED POWER

A steady state reactor core heat output of 2546 MWt.
e
2587

B. THERMAL POWER

The total core heat transferred from the fuel to the coolant.

C. REACTOR OPERATION

1. REFUELING SHUTDOWN

When the reactor is subcritical by at least 5% $\Delta k/k$ and T_{avg} is $\leq$ 140°F and fuel is scheduled to be moved to or from the reactor core.

2. COLD SHUTDOWN

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq$ 200°F.

3. INTERMEDIATE SHUTDOWN

When the reactor is subcritical by at least 1.77% $\Delta k/k$ and 200°F < T_{avg} < 547°F.

4. HOT SHUTDOWN

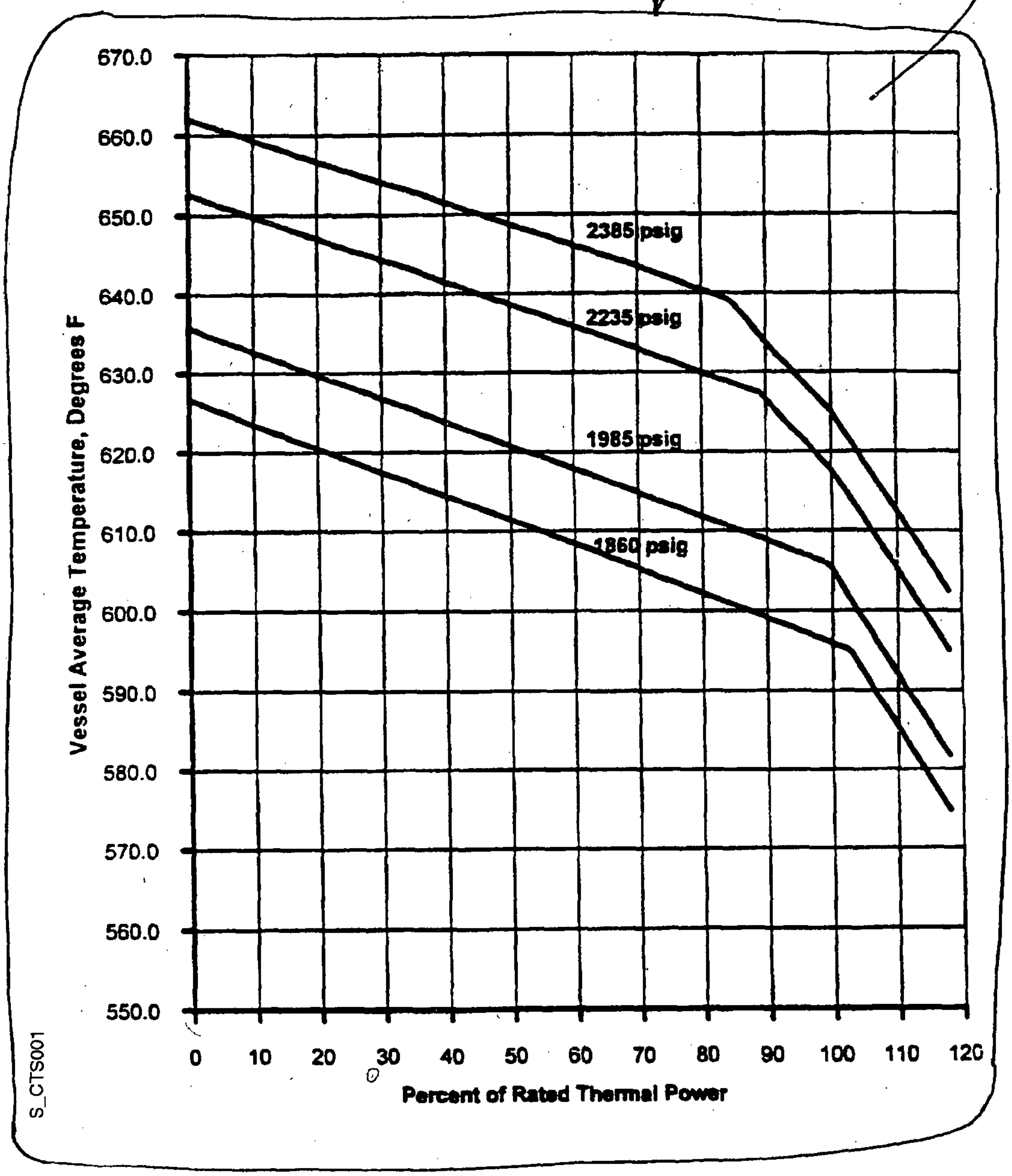
When the reactor is subcritical by at least 1.77% $\Delta k/k$ and T_{avg} is $\geq$ 547°F.

08-03-95 e

TS FIGURE 2.1-1

REACTOR CORE THERMAL AND
HYDRAULIC SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW

INSERT TS FIGURE 2.1-1



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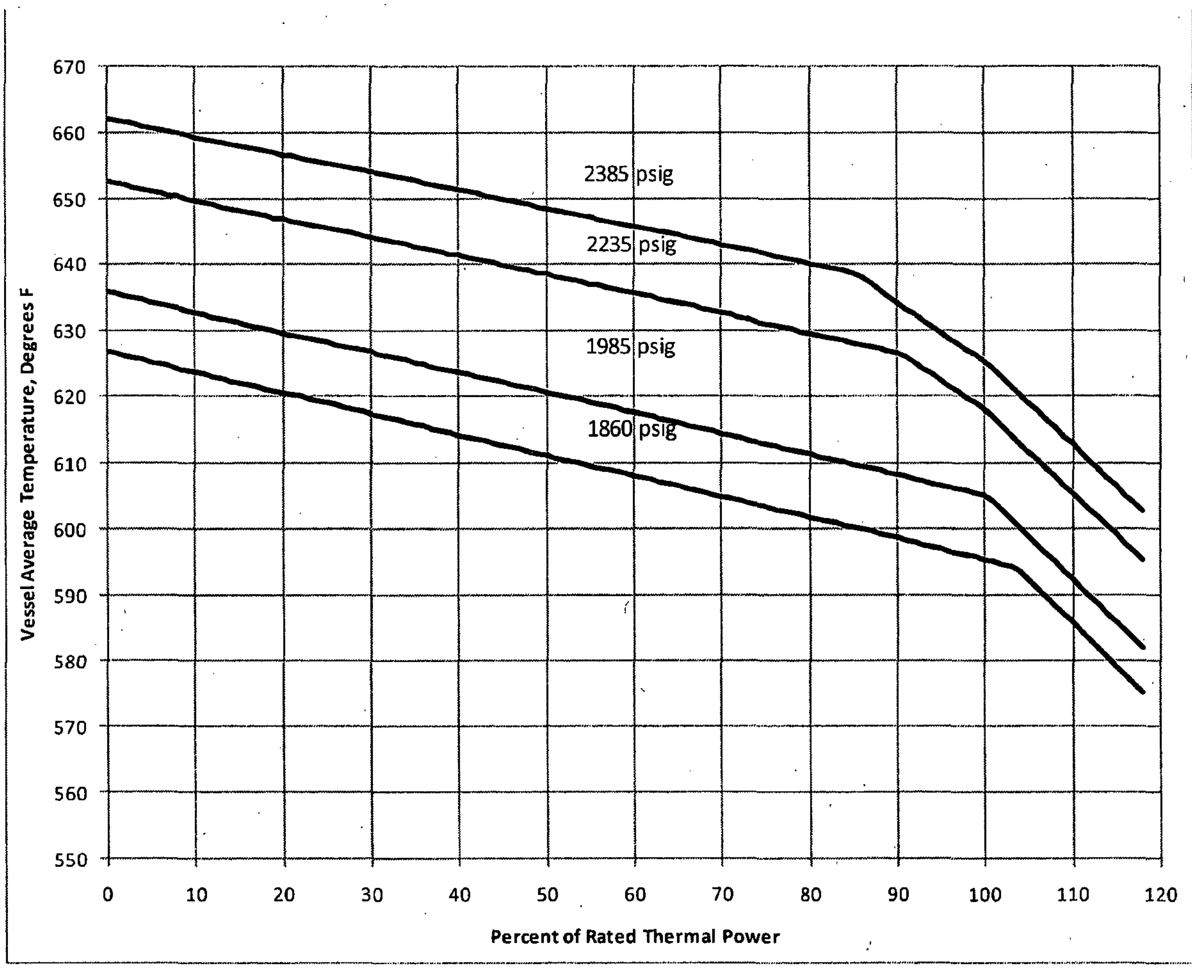
TS FIGURE 2.1-1



(b) High pressurizer pressure - $\leq$ 2380 psig.

(c) Low pressurizer pressure - $\geq$ 1875 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0\left[K_1 - K_2\left(\frac{1 + t_1 s}{1 + t_2 s}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

where

ΔT_0 = Indicated ΔT at rated thermal power, °F
T = Average coolant temperature, °F
T' = 573.0°F
P = Pressurizer pressure, psig
P' = 2235 psig
K_1 = 1.135
K_2 = 0.01072
K_3 = 0.000566 0.000770
$\Delta I = q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power
$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1
$t_1 \geq$ 29.7 seconds
$t_2 \leq$ 4.4 seconds

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% ΔT Power.)

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0\left[K_4 - K_5\left(\frac{t_3 s}{1 + t_3 s}\right)T - K_6(T - T') - f(\Delta I)\right]$$

ATTACHMENT 3

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES (TYPED)

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

   B. Technical Specifications

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

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

B. <u>Technical Specifications</u>

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

C. <u>Reports</u>

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. <u>Records</u>

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Deleted by Amendment 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

1.0 DEFINITIONS

The following frequently used terms are defined for the uniform interpretation of the specifications.

A. RATED POWER

A steady state reactor core heat output of 2587 MWt.

B. THERMAL POWER

The total core heat transferred from the fuel to the coolant.

C. REACTOR OPERATION

1. REFUELING SHUTDOWN

When the reactor is subcritical by at least 5% $\Delta k/k$ and T_{avg} is $\leq 140°$ F and fuel is scheduled to be moved to or from the reactor core.

2. COLD SHUTDOWN

When the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is $\leq 200°$ F.

3. INTERMEDIATE SHUTDOWN

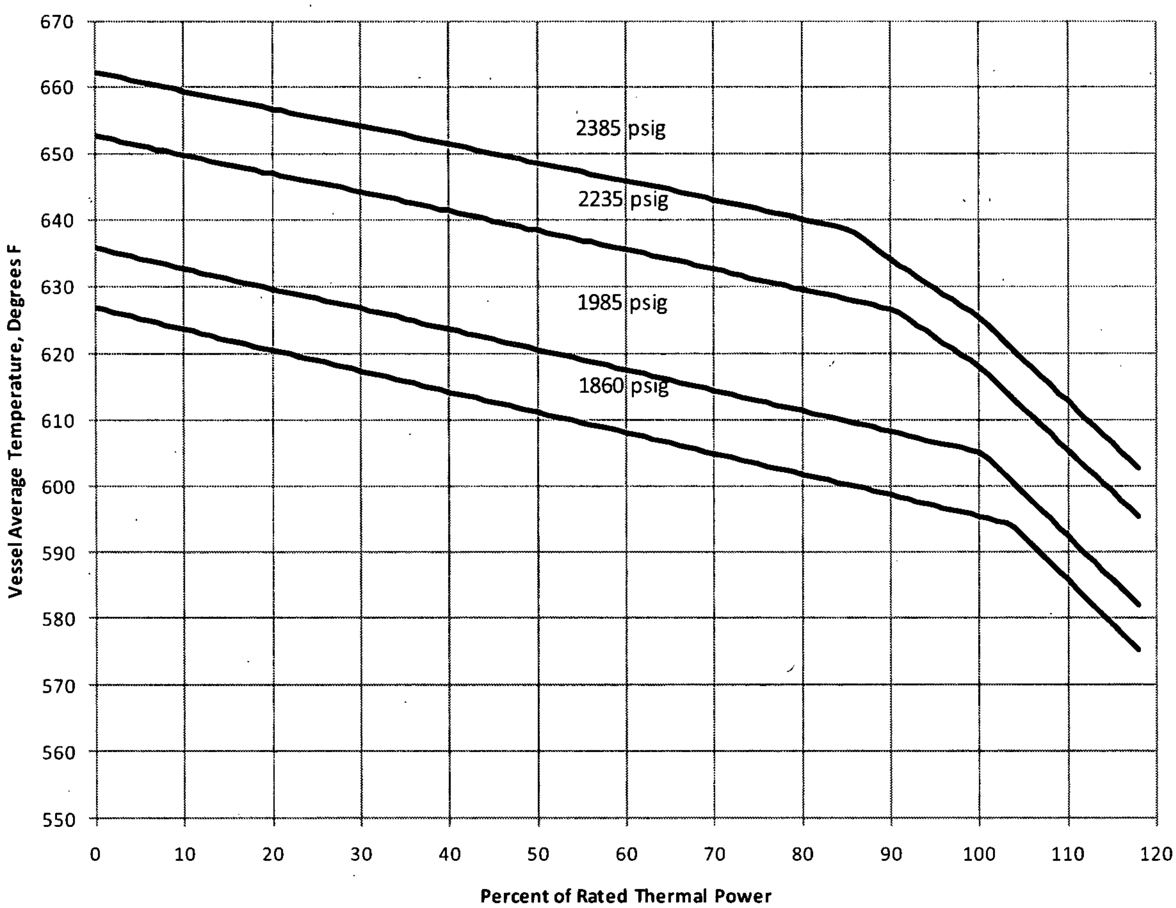
When the reactor is subcritical by at least 1.77% $\Delta k/k$ and $200°\text{ F} < T_{avg} < 547°$ F.

4. HOT SHUTDOWN

When the reactor is subcritical by at least 1.77% $\Delta k/k$ and T_{avg} is $\geq 547°$ F.

TS FIGURE 2.1-1

REACTOR CORE THERMAL AND
HYDRAULIC SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW



Amendment Nos.

(b) High pressurizer pressure - ≤2380 psig.

(c) Low pressurizer pressure - ≥ 1875 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_0\left[K_1 - K_2\left(\frac{1 + t_1 s}{1 + t_2 s}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

where

ΔT_0 = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = 573.0°F

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.135

K_2 = 0.01072

K_3 = 0.000770

ΔI = q_t - q_b, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1

$t_1 \geq 29.7$ seconds

$t_2 \leq 4.4$ seconds

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of the ΔT span. (Note that 2.0% of the ΔT span is equal to 3.0% ΔT Power.)

(e) Overpower ΔT

$$\Delta T \leq \Delta T_0\left[K_4 - K_5\left(\frac{t_3 s}{1 + t_3 s}\right)T - K_6(T - T') - f(\Delta I)\right]$$

ATTACHMENT 4

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

PROPOSED TECHNICAL SPECIFICATIONS BASES AND TECHNICAL REQUIREMENTS MANUAL PAGES (MARKED-UP)

FOR INFORMATION ONLY

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits as specified in the CORE OPERATION LIMITS REPORT ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 573.0°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

+0.38%

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also

The specified minimum water volume in the 110,000-gallon protected condensate storage tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all offsite electrical power. It is also sufficient to maintain one unit at hot shutdown for 2 hours, followed by a 4 hour cooldown from 547°F to 350°F (i.e., RHR operating conditions). If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000-gallon condensate tank can be gravity-fed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,842,454 pounds per hour at their individual relieving pressure; the total combined capacity of all fifteen main steam code safety valves is 11,527,362 pounds per hour. The nominal power rating steam flow is 11,260,000 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady state power than can be obtained during three reactor coolant loop operation. maximum at full power approximately 11,444,000

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine-131 activity is based on limiting the inhalation dose at the site boundary following a postulated steam line break accident to a small fraction of the 10 CFR 100 limits. The accident analysis, which is performed based on the guidance of NUREG-0800 Section 15.1-5, assumes the release of the entire contents of the faulted steam generator to the atmosphere.

3.3 INSTRUMENTATION

3.3.5 Feedwater Ultrasonic Flow Meter Calorimetric

TR 3.3.5 The Feedwater Ultrasonic Flow Meter (UFM) Calorimetric shall be FUNCTIONAL with:

a. The Feedwater UFM System FUNCTIONAL.

b. The Plant Computer System (PCS) calorimetric program FUNCTIONAL.

APPLICABILITY: Power Operations with THERMAL POWER > 2546 MWt (98.4% rated power).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Feedwater UFM System not FUNCTIONAL.	A.1 Change the calorimetric program from the Feedwater UFM System to the Normalized Feedwater Venturi System.	1 hour
	<u>AND</u>	
	A.2 Restore Feedwater UFM System to FUNCTIONAL status.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to ≤2546 MWt (98.4% rated power).	1 hour
	<u>AND</u>	
	B.2 Change the calorimetric program from the Normalized Feedwater Venturi System to the Feed or Steam Venturi System.	1 hour

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. PCS calorimetric program not FUNCTIONAL for reasons other than Condition A.	C.1 Verify THERMAL POWER ≤2587 MWt (100% rated power) by monitoring alternate power indications.	Immediately
	<u>AND</u>	
	C.2.1 Restore the PCS calorimetric program to FUNCTIONAL status.	Prior to performing the next required power range channel calorimetric heat balance comparison per TS Table 4.1-1 item 1
	<u>OR</u>	
	C.2.2 Reduce THERMAL POWER to ≤2546 MWt (98.4% rated power) by monitoring alternate power indications.	Prior to performing the next required power range channel calorimetric heat balance comparison per TS Table 4.1-1 item 1

TRM SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.5.1	-------------------NOTE-------------------- When in Condition A, the Normalized Feedwater Venturi System will be used to perform the daily surveillance. --- Perform heat balance IAW TS Table 4.1-1 item 1 using the Feedwater UFM System.	Prior to exceeding 2546 MWt (98.4% rated power) <u>AND</u> Daily thereafter
TSR 3.3.10.2	Perform Channel Calibration of the Feedwater UFM System instrumentation.	Once per 18 months

B 3.3 INSTRUMENTATION

B 3.3.5 Feedwater Ultrasonic Flow Meter Calorimetric

BASES

BACKGROUND AND APPLICABLE SAFETY ANALYSES

Surry was initially licensed to operate at a maximum reactor power level of 2441 (MWt). In August 1995, the Nuclear Regulatory Commission (NRC) approved SPS operation at the reactor power level of 2546 MWt. A second power uprate to a reactor power level of 2587 MWt is based on a redistribution of analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed per the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models." Appendix K originally mandated 102% of licensed power level for light water reactor ECCS evaluation models. The NRC approved a change to the 10 CFR 50, Appendix K requirements on June 1, 2000. This change provided licensees the option of maintaining the 2% power margin between the licensed power level and the ECCS evaluation assumed power level, or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The Feedwater Ultrasonic Flow Meter (UFM) System provides a more accurate measurement of feedwater flow compared to the feedwater venturis and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty that is less than the accident analysis allowance of 0.38% rated power. The remaining approximately 1.6% rated power margin is the basis for the power uprate. This type of power uprate is referred to as a Measurement Uncertainty Recapture (MUR) Uprate.

The Feedwater UFM System provides on-line main feedwater flow and temperature measurements to determine reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The system measures the transit times of ultrasonic pulses traveling through the flowing fluid. Sound travels faster when the pulse traverses the pipe with the flow and slower against the flow due to the doppler effect. The system uses these transit times and time differences between pulses to

(continued)

BASES

BACKGROUND AND APPLICABLE SAFETY ANALYSES (continued)	determine the fluid velocity. The system also measures the speed of sound in water and uses this measurement to determine the feedwater temperature. The Feedwater UFM System consists of an electronic processing cabinet installed in the Cable Spreading Room, and measurement spool pieces installed in each of the three main Feedwater flow lines. Each measurement spool piece consists of 16 ultrasonic, multi-path, transit time transducers, one dual resistance temperature detector (RTD), and two pressure transmitters. The 16 transducers are separated into two planes, four paths in each plane. Each plane provides input to its own subsystem of electronic hardware. Each transducer can be removed without disturbing the pressure boundary. The electronics for the two subsystems, while electrically separated, are housed in a single processing cabinet installed in the Cable Spreading Room. The Feedwater UFM System performs on-line self diagnostics to verify system operation within design basis uncertainty limits. Any out of specification condition will result in a control room annunciator. A failure between the Feedwater UFM System electronics cabinet and the plant computer will also result in a control room annunciator. If the feedwater UFM failure annunciator is received, the Feedwater UFM System will need to be declared not functional. The control room annunciator response procedure provides guidance to the operators for initial alarm diagnosis and response. Although a single plane malfunction results in a minimal increase in feedwater flow uncertainty, operators will conservatively respond to a single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode. The Feedwater UFM System will be used in place of the Steam or Feed flow instrumentation to perform the calorimetric calculation. The venturi based flow instruments will continue to provide inputs to other indications, protection and control systems, and will be used if the Feedwater UFM System is not available.
LCO	The Feedwater UFM calorimetric requires the Feedwater UFM System and the Plant Computer System (PCS) calorimetric program to be FUNCTIONAL.

BASES

APPLICABILITY	During Power Operations with THERMAL POWER > 2546 MWt (98.4% rated power), the Feedwater UFM calorimetric must be FUNCTIONAL. The Feedwater UFM calorimetric provides a more accurate measurement of reactor thermal power than the steam or feed flow venturi-based calorimetric. The improved accuracy of the Feedwater UFM calorimetric is the basis for operating above 2546 MWt (98.4% rated power).

ACTIONS

A.1 and A.2

With the Feedwater UFM System nonfunctional, action must be taken to restore FUNCTIONAL status in 48 hours, provided THERMAL POWER remains above 2546 MWt (98.4% rated power). If the Feedwater UFM System is not returned to service in 48 hours, reactor power is required to be reduced to ≤2546 MWt (98.4% rated power).

The Normalized Feedwater Venturi System calorimetric is used during the 48 hour completion time when the Feedwater UFM System is nonfunctional. The Normalized Feedwater Venturi System calorimetric receives input from the feedwater venturis for feedwater flow rate calculation. The feedwater flow from the three venturis is normalized to the Feedwater UFM System flow rate. In addition, the feedwater temperature and feedwater pressure data is normalized to the more accurate data from the Feedwater UFM System. Normalization of data results in the Normalized Feedwater Venturi System calorimetric closely matching the Feedwater UFM System calorimetric.

The accuracy of the instruments used to perform the Normalized Feedwater Venturi System calorimetric will not significantly change over 48 hours. As a result, significant changes in calorimetric measurement uncertainty will not occur over a 48 hour period. The 1 hour completion time to change the calorimetric program from the Feedwater UFM System to the Normalized Feedwater Venturi System is reasonable based on operating experience.

During the 48-hour COMPLETION TIME, if THERMAL POWER is reduced to ≤2546 MWt (98.4% rated power), THERMAL POWER cannot be increased to > 2546 MWt (98.4% rated power) until a calorimetric is performed using the Feedwater UFM System in accordance with TS Table 4.1-1, item 1. This restriction is required to ensure that the plant transient has not affected the accuracy of the Normalized Feedwater Venturi System.

BASES

ACTIONS (continued)

B.1 and B.2

If the Feedwater UFM System cannot be restored to functional status within the associated completion time, the unit must be placed in a condition in which the LCO does not apply. To achieve this status, reactor power must be reduced to ≤2546 MWt (98.4% rated power). The 1 hour completion time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The accuracy of the Normalized Feedwater Venturi calorimetric program can be impacted over time. For this reason, the calorimetric program will be changed from the Normalized Feedwater Venturi System to the Feed or Steam Venturi System after a 48-hour time period. The 1-hour completion time to change the calorimetric program from the Normalized Feedwater Venturi System to the Feed or Steam Venturi System is reasonable based on operating experience.

C.1, C.2.1 and C.2.2

A failure of the PCS calorimetric program would result in the loss of computer generated calorimetric programs. In this case, THERMAL POWER would be determined by monitoring alternate power indications using the power range nuclear instrumentation (NIs) and RCS loop ΔTs. The procedure for loss of the PCS provides guidance for monitoring reactor power.

Operation at 100% rated power may continue until the next required performance of TS Table 4.1-1, item 1. If the computer calorimetric program is nonfunctional, a manual calorimetric heat balance calculation would be required to meet the requirements of TS Table 4.1-1, item 1. The manual calorimetric heat balance calculation uses data from the feedwater venturis, not the Feedwater UFM System. Therefore, the manual calorimetric cannot be used to satisfy the surveillance requirement when operating above 2546 MWt (98.4% rated power).

If the PCS calorimetric program is not restored to FUNCTIONAL status prior to the performance of the next calorimetric required by TS Table 4.1-1, item 1, THERMAL POWER would be reduced to ≤2546 MWt (98.4% rated power) and a manual calorimetric would be performed. The power reduction and performance of a manual calorimetric would

(continued)

BASES

ACTIONS

C.1, C.2.1 and C.2.2 (continued)

have to be completed within the surveillance interval required by TS Table 4.1-1, item 1. Thermal power would be reduced by monitoring alternate power indications using the power range nuclear instrumentation (NIs) and RCS Loop ΔTs.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

Note 1 has been added to clarify that when in Condition A, the Normalized Feedwater Venturi System will be used to perform the required daily surveillance.

This SR ensures that a calorimetric using the more accurate measurements of feedwater flow from the Feedwater UFM System is performed prior to exceeding a THERMAL POWER level of 2546 MWt (98.4% rated power). The Feedwater UFM System is used to perform the TS Table 4.1-1, item 1 surveillance daily thereafter.

If THERMAL POWER is reduced to ≤2546 MWt (98.4% rated power), a calorimetric using the Feedwater UFM System must be performed prior to exceeding 2546 MWt (98.4% rated power). This initial surveillance is required to be performed even if power is reduced for a short period of time and a calorimetric using the Feedwater UFM System had been performed within the previous daily requirement.

A calorimetric using the Feedwater UFM System is required to be performed each time power will be increased > 2546 MWt (98.4% rated power). This ensures the requirements (feedwater UFM calorimetric) are met for operating at a power level of > 2546 MWt (98.4% rated power).

A channel calibration of the Feedwater UFM System instrumentation is performed every 18 months. The Feedwater UFM System instrumentation calibration procedure and frequency of calibration, are based on vendor recommendations.

REFERENCES

1. License Amendment Request – Measurement Uncertainty Recapture Power Uprate (LBDCR No. TSCR 409 submitted to NRC via Letter Serial No. 09-223).
2. TS Table 4.1-1, item 1.

ATTACHMENT 5

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

NRC REGULATORY ISSUE SUMMARY 2002-03
REQUESTED INFORMATION

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

Table of Contents

Acronym List

Expression	Definition or Use
AC	alternating current
AFW	auxiliary feedwater
AMSAC	anticipated transient without scram (ATWS) mitigation system actuation circuitry
AOR	analysis of record
AOV	air operated valve
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BELOCA	Best-Estimate loss-of-coolant accident
BOP	balance-of-plant
B&PV	boiler and pressure vessel
CCW	component cooling water
CFR	Code of Federal Regulations
CPU	central processing unit
CRDM	control rod drive mechanism
CS	containment spray
CUF	cumulative usage factor
CW	circulating water
DBA	design basis accident
DC	direct current
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOR	Division of Operating Reactors
EAB	exclusion area boundary
ECCS	emergency core cooling system
ECST	emergency condensate storage tank
EDG	emergency diesel generator
EFPY	effective full power years
EOL	end of life
EOP	emergency operating procedure
EQ	environmental qualification (10 CFR 50.49)
ESF	engineered safety features

Acronym List

Expression	Definition or Use
ESFAS	engineered safety features actuation system
FAC	flow accelerated corrosion
FW	feedwater
GL	NRC Generic Letter
HELB	high energy line break
HFP	hot full power
HP	high pressure
HZP	hot zero power
ISI	inservice inspection
IST	inservice testing
LAR	license amendment request
LEFM	leading edge flow meter
LOCA	loss of coolant accident
LOFA	loss of flow accident
LOL	loss of external electrical load
LOOP	loss of offsite power
LPZ	low population zone
LRA	locked RCP rotor/sheared shaft accident
LTOPS	low temperature overpressure protection system
MDNBR	minimum departure from nucleate boiling ratio
MFLB	major feedwater line break
MFO	maximum facility output
MOV	motor operated valve
MSLB	main steam line break
MSS	main steam system
MSSV	main steam safety valve
MSTV	main steam trip valve
MSVH	main steam valve house
MUR	measurement uncertainty recapture
MWD/MTU	megawatt day per metric ton uranium
MWe	megawatt electric
MWt	megawatt thermal
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system

Acronym List

Expression	Definition or Use
OPΔT	overpower delta temperature
OTΔT	overtemperature delta temperature
PCT	peak clad temperature
PCS	plant computer system
PORV	power operated relief valve
PSV	pressurizer safety valve
PTS	pressurized thermal shock (10 CFR 50.61)
PWR	pressurized water reactor
RCCA	rod control cluster assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RCSL	reactor core safety limits
RHR	residual heat removal
RP	rated power
RPS	reactor protection system
RPV	reactor pressure vessel
RS	recirculation spray
RT_{NDT}	reference temperature nil ductility transition
RT_{PTS}	reference temperature pressurized thermal shock
RTD	resistance temperature detector
RVID	reactor vessel integrity database
RWAP	rod withdrawal at power
RWSC	Rod Withdrawal from Subcritical
RWST	refueling water storage tank
SBLOCA	small break loss-of-coolant accident
SBO	station blackout
SER	safety evaluation report
SFP	spent fuel pool
SG	steam generator
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SIF	Surry Improved Fuel
SPS	Surry Power Station
SSC	system, structure, or component

Acronym List

Expression	Definition or Use
SUIL	Start-up of an Inactive Loop
SW	service water
TEDE	total effective dose equivalent
TRM	Technical Requirements Manual
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UFM	ultrasonic flow meter
VCT	volume control tank
VEPCO	Virginia Electric and Power Company
WGDT	Waste Gas Decay Tank
χ/Q	radiological atmospheric dispersion factor

Introduction

This attachment contains the Dominion responses to the NRC Regulatory Issue Summary 2002-03, requested information for MUR power uprates. The LAR attachment sections match the NRC Regulatory Issue Summary 2002-03 sections for ease of review.

I FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

1. A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. The description should include:

   A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

   B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

   C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

   D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

   E. A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty

   F. Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

      i. maintaining calibration

      ii. controlling software and hardware configuration

      iii. performing corrective actions

      iv. reporting deficiencies to the manufacturer

v. receiving and addressing manufacturer deficiency reports

G. A proposed allowed outage time for the instrument, along with the technical basis for the time selected

H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

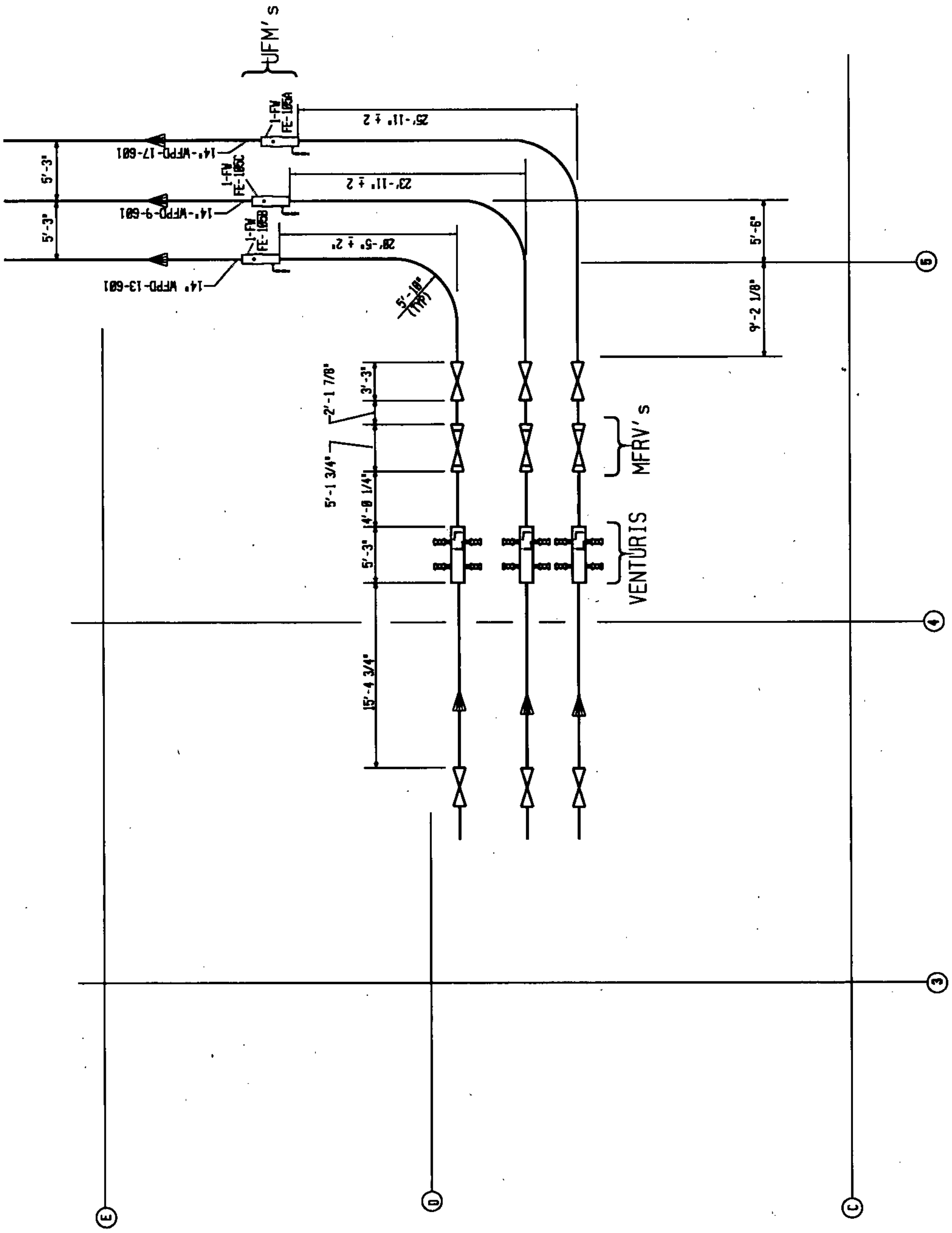
RESPONSE TO I – FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

I.1 Detailed Description of the Surry Units 1 and 2 Implementation of the Feedwater Ultrasonic Flow Meter

The SPS feedwater ultrasonic flow meter is a Cameron LEFM CheckPlus ultrasonic multi-path, transit time flowmeter. This equipment also provides a highly accurate feedwater temperature that will be input to the heat balance. This advanced flow measurement system design is described in detail by the manufacturer, Cameron Inc. (formerly Caldon), in Topical Reports ER-80P, Revision 0 (Reference I-1), and ER-157P, Revision 5 (Reference I-2).

The LEFM CheckPlus system consists of an electronic cabinet installed in the Cable Spreading Room, and measurement spool pieces installed in each of the three main feedwater flow lines between the existing feedwater venturi flow meters and the main feedwater check valves. The spool pieces are installed well downstream of the existing feedwater flow venturis, and will have no impact on venturi performance. The FW piping is 14-inch, Schedule 80 pipe with a nominal inside diameter of 12.5 inches. Cameron Installation and Commissioning Manual (IB0712), Section 1.1.1, requires at least five inside pipe diameters downstream of the centerline of an upstream disturbance. The spool piece metering sections are installed a minimum of 19 inside pipe diameters downstream of the FW flow venturis. Figures I-1 and I-2 provide details of the UFM location for Units 1 and 2. The installation of the spool piece metering sections will create less than 0.015 psi of additional head loss in the feedwater system. Therefore, the impact on the venturis is insignificant. The UFMs were calibrated at the Alden Research Laboratory facility using the current plant piping configuration and variations of the plant configuration. The calibration test determines the meter calibration constant or meter factor. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests are performed to determine meter factor sensitivity to upstream hydraulics.

Figure I-1
Detail from Surry Unit 1 Station Control Drawing 11448-FP-2C



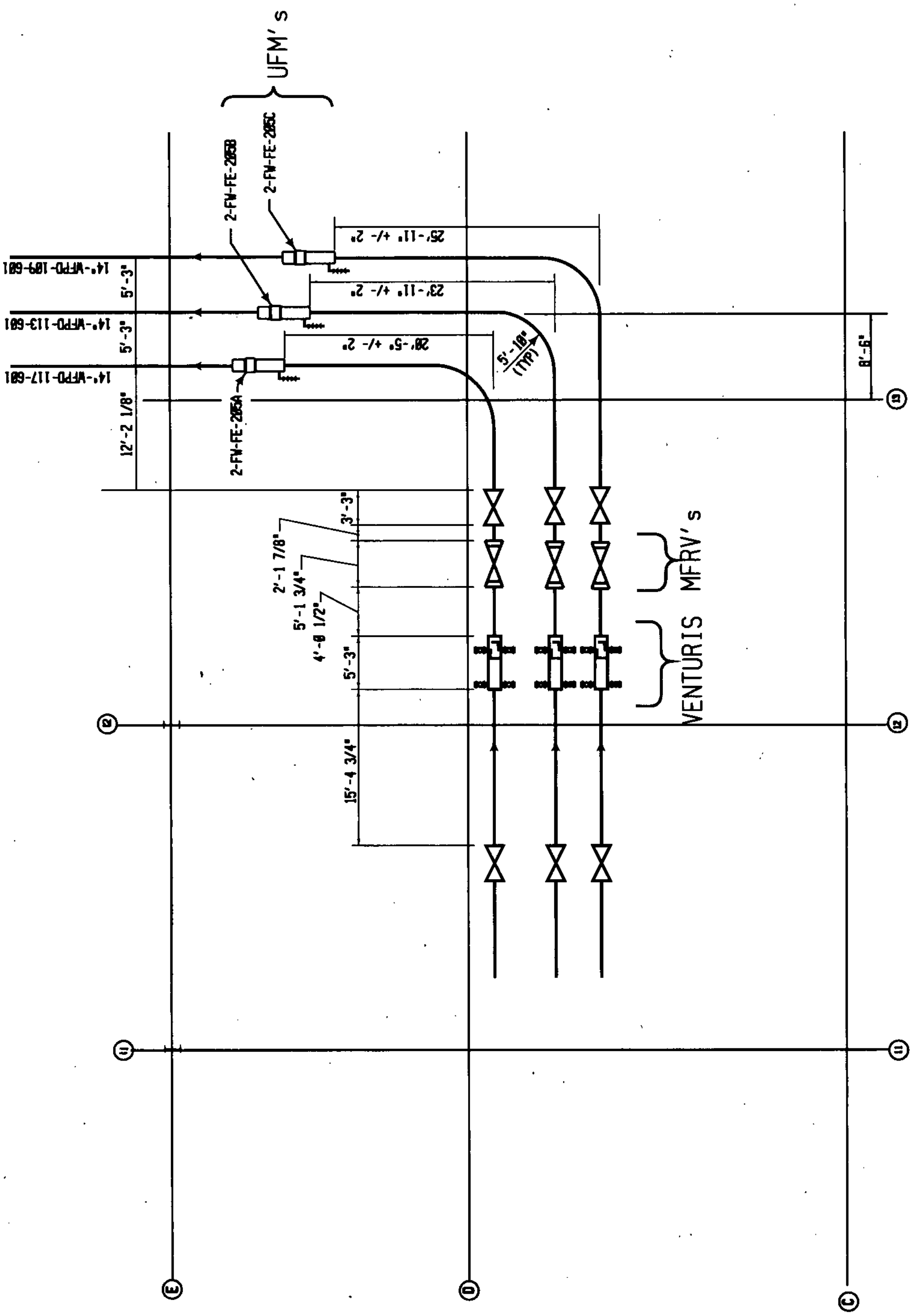


Figure I-2
Details Created from Surry Unit 2 Project Drawing S-08008-2-2FP2C-B

Each measurement section consists of 16 ultrasonic, multi-path, transit time transducers, one dual RTD, and two pressure transmitters. Each transducer may be removed at full power conditions without disturbing the pressure boundary. These flow elements conform to the installation location requirements specified in Topical Reports ER-80P and ER-157P. Station procedures for transducer replacement will be developed based on Cameron Engineering Field Procedure 18, *Installation Procedure for In-Line Pushrod Transducer.*

The UFM measures the transit times of ultrasonic energy pulses traveling along chordal acoustic paths through the flowing fluid. This technology provides higher accuracy and reliability than the existing flow instruments. Sound travels faster when the pulse traverses the pipe with the flow and slower against the flow due to the Doppler effect. The UFM uses these transit times and time differences between pulses to determine the fluid velocity. The UFM also measures the speed of sound in water and uses this measurement to determine the feedwater temperature.

The electronic cabinet controls magnitude and sequences transducer operation; makes time measurements; and calculates volume, temperature and mass flow. The system software employs the ultrasonic transit time method to measure velocities at precise locations. The system numerically integrates the measured velocities. The system software has been developed and maintained under a verification and validation program. The verification and validation program has been applied to system software and hardware, and includes a detailed code review. The feedwater mass flow rate and temperature are displayed on the electronic cabinet and transmitted to the plant process computer for use in the calorimetric measurement (secondary plant energy balance) of reactor thermal output. The system will utilize continuous calorimetric power determination by direct, redundant links with the plant computer, and will incorporate self-verification features. These features ensure that system performance is consistent with the design basis.

The system has two modes of operation: Normal operation and Maintenance mode. Normal operation is defined as CheckPlus operation. In this mode, both planes of transducers are in service and system operations are processed by both CPUs. If the system is subjected to a failure involving a transducer, failure of one plane of operation or if a CPU related malfunction occurs, the system reverts to the Check system or Maintenance mode. When a plane of operation is lost, the system alerts the control room operators through the annunciator window for Feedwater Ultrasonic Flow Meter Failure, and shifts from Normal operation to Maintenance mode. If the system suffers a loss of AC power or other total failure, the system also alerts the operators through the aforementioned annunciator. Operations personnel are also alerted to system trouble through the annunciator window for Feedwater Ultrasonic Flowmeter Trouble if the electronic cabinet internal temperature is high or when other trouble conditions occur as determined by the plant computer.

The improved measurement accuracy for feedwater mass flow and temperature results in a total uncertainty of 0.35% of RP. This is more accurate than the nominal 2% RP used in the accident analyses or the uncertainty currently obtainable with precision, venturi-based instrumentation and RTDs.

The UFM indications of feedwater mass flow and temperature will be directly substituted for the existing venturi-based flow and RTD temperature inputs currently used in the plant calorimetric measurement calculations. The plant computer system calorimetric programs will be revised to receive data from the UFM and from loop-specific, high-capacity SG blowdown flow, to calculate UFM mode calorimetric power, and to calculate feedwater and steam venturi flow calorimetric power using feedwater flow, temperature and pressure values normalized to UFM values. The existing venturi-based feedwater flow and RTD temperature will continue to be used for feedwater control and other functions, and may be used for plant calorimetric measurement calculations when the UFM is unavailable.

I.1.A Cameron Topical Reports Applicable to the LEFM CheckPlus System

The referenced Topical Reports are:

ER-80P, Rev. 0 (Reference I-1)

ER-157P, Rev. 5 (Reference I-2)

I.1.B NRC Approval of Cameron LEFM CheckPlus System Topical Reports

The NRC approved the Topical Reports referenced in I.1.A above on the following dates:

ER-80P, NRC SER dated March 8, 1999 (Reference I-3)

ER-157P, NRC SER dated December 20, 2001 (Reference I-4)

The NRC performed additional evaluations on the acceptability of the Cameron LEFM. The evaluation results are documented in Reference I-5, which addressed the hydraulic aspects of Cameron LEFMs in response to industry operating experience. The NRC staff concluded that the Cameron Check and CheckPlus LEFM performance was consistent with the Cameron Topical Reports ER-80P, Revision 0 and ER-157P, Revision 5, previously approved by the NRC staff (Reference I-5).

I.1.C Surry Power Station (SPS) Implementation of Guidelines and NRC SER for the Cameron LEFM CheckPlus System

The LEFM CheckPlus system is permanently installed per the requirements specified in Topical Reports ER-80P and ER-157P. The system will be used for continuous calorimetric power determination by direct, redundant links with the plant computer. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The plant computer system software continuously adjusts the venturi flow coefficients and the feedwater RTD temperatures to the more accurate UFM values. The feedwater flow values for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow are normalized to equal the UFM feedwater flow. Normalization is performed on a loop basis. The feedwater temperature values used to determine densities for the new normalized filtered feedwater venturi flow and normalized one minute average feedwater venturi flow are based on normalized feedwater RTD temperatures biased to equal the UFM feedwater temperatures. Feedwater pressure measurements will be normalized to the more accurate data from the UFM.

The SPS LEFM CheckPlus system was calibrated in a site-specific model test at Alden Research Labs with traceability to National Standards. A copy of the Alden Research Labs certified calibration report is contained in the Cameron Meter Factor Reports. The LEFM CheckPlus system installation and commissioning is performed according to Cameron procedures. These procedures include verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those during site-specific model testing.

I.1.D Disposition of NRC SER Criteria During Installation

In approving Cameron Topical Reports ER-80P and ER-157P, the NRC established four criteria each licensee must address. The four criteria are listed below along with a discussion of how each will be satisfied for SPS Units 1 and 2.

I.1.D.1 NRC Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the UFM. These procedures should include processes and contingencies for a not functional UFM and the effect on thermal power measurement and plant operation.

I.1.D.1.1 Response to NRC Criterion 1

Power uprate license amendment implementation will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing and training at the uprated power level using the new LEFM CheckPlus system. A preventive maintenance program will be developed for the UFM using the vendor's maintenance and troubleshooting manual. Work on the UFM will be performed by site instrumentation and control personnel qualified per the SPS Instrumentation & Control Training Program.

The preventive maintenance activities include:

- General terminal and cleanliness inspection
- Power supply inspection
- CPU inspection
- Acoustic Processor Unit checks
- Analog input/output checks
- Alarm Relay checks
- Watchdog Timer checks that ensure the software is running
- Communication checks
- Transducer checks
- Calibration checks on each feedwater pressure transmitter

The preventive maintenance program and UFM continuous self-monitoring ensure that the UFM remains bounded by the Topical Report ER-80P analysis and assumptions. Establishing and continued adherence to these requirements assures that the UFM system is properly maintained and calibrated.

Contingency plans for plant operation with the UFM not functional are described in Sections I.1.G and I.1.H below.

I.1.D.2 NRC Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system, and bounds the analysis and assumptions set forth in Topical Report ER-80P.

I.1.D.2.1 Response to NRC Criterion 2

UFM spool pieces were installed in Unit 1 during the Spring 2009 Refueling Outage and Unit 2 during the Fall 2009 Refueling Outage. Commissioning and calibration is expected to be completed by April 2010. Monitoring will be initiated following installation. The UFM feedwater flow and temperature data will be compared to the feedwater flow venturis output and the feedwater RTD output. The data comparison is expected to demonstrate that the UFM is consistent with the venturi feedwater flow and RTD feedwater temperature. There have been no maintenance related activities since LEFM installation.

I.1.D.3 NRC Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and the LEFM for comparison.

I.1.D.3.1 Response to NRC Criterion 3

Dominion uses a core thermal power uncertainty calculation approach consistent with ISA-RP67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation* (Reference I-14); and Cameron's Topical Report ER-80P (Reference I-1), as supplemented by ER-157P (Reference I-2). The combination of errors within instrument loops is accomplished per Dominion Technical Report EE-0116 (Reference I-15). This document is referenced in the Technical Specification 3.7 Basis. An alternate methodology for calculating UFM uncertainty was not used.

The fundamental approach used in the setpoint methodology is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then combined using the square root of the sum of the squares approach to determine the overall uncertainty. The same fundamental approach was used to determine the UFM based power calorimetric uncertainty. This approach has been approved by the NRC in Cameron Topical Reports ER-80P and ER-157P as well as for Seabrook Station Unit 1 (Reference I-11), Vogtle Electric Generating Plant (Reference I-12), and Cooper Nuclear Station (Reference I-13).

I.1.D.4 NRC Criterion 4

For plants where the LEFM was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), provide additional justification for use. This justification should show either that the meter installation is independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher

Reynolds numbers. Additionally, for previously installed and calibrated LEFM, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

I.1.D.4.1 Response to NRC Criterion 4

A UFM bounding uncertainty has been provided for use in the uncertainty calculation described below (References I-6 and I-7). The bounding calibration factor acceptability for the SPS spool pieces was established by tests at the Alden Research Labs (References I-8 and I-9). These tests included a full-scale model of the Unit 1 and 2 hydraulic geometry and a straight pipe. An Alden Research Labs test data report and Cameron engineering report evaluating the test data were prepared. The calibration factor used for the UFM is based on these reports. The spool piece calibration factor uncertainty is based on the Cameron engineering reports. The site specific uncertainty analysis documents these analyses and will be maintained as part of the SPS technical basis for the power uprate.

Final site-specific uncertainty analyses acceptance will occur after completion of the commissioning process. The commissioning process verifies bounding calibration test data and provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. Final commissioning is expected to be completed by March 2010.

I.1.E Total Power Measurement Uncertainty at Surry Units 1 and 2

The overall thermal power uncertainty using the UFM is 0.35% at RP for Units 1 and 2. The uncertainty calculations for Surry Units 1 and 2 are documented in References I-6 and I-7, which are Cameron proprietary documents that will be transmitted to the NRC via separate proprietary letter from Dominion. The key parameters and their uncertainty are summarized in Table I-1. In addition to the calorimetric inputs provided by the UFM System for determination of feedwater mass flow rate and enthalpy, the Surry plant computer uses several process inputs (e.g., charging flow, letdown flow, steam generator blowdown flow) to calculate the contribution of steam enthalpy and other gains and losses that are identified as Items 7 and 8 in Table I-1. For comparison, baseline values from Cameron ER-157P, Revision 5 (Reference I-2) are presented in Table I-1. Differences between the Surry uncertainties and those from ER-157P, Revision 5 are a result of plant-specific calculations and parameter uncertainties.

In regard to Item 7 in Table I-1, Surry Units 1 and 2 do not use zero-moisture steam. A steam moisture carryover (MCO) uncertainty of 0.08% moisture is included in the power calorimetric uncertainty calculation. Using this MCO uncertainty, a steam moisture uncertainty of 0.042% at RP was developed. This value was combined with a steam pressure uncertainty of 0.064% RP to develop a plant-specific uncertainty for steam enthalpy of 0.077% RP (presented as 0.08%

in Table I-1 of the License Amendment Request). The steam moisture and pressure uncertainties are Items 20 and 21, respectively, in Table B-1 of ER-651, Revision 1 (Unit 2) and ER-650, Revision 2 (Unit 1).

The uncertainty for transducer installation, as identified in Cameron Customer Information Bulletin CIB-125 (Reference I-10), has been included in the UFM System uncertainty for Surry Unit 1 (Reference I-6) and Surry Unit 2 (Reference I-7). These system uncertainties incorporate an additional transducer variability uncertainty in both the profile factor uncertainty and in the installation uncertainty.

Table I-1
Total Thermal Power Uncertainty Determination for Surry Units 1 and 2

Item	Parameter[1]	ER-157P, Rev. 5 Uncertainty	Unit 1 Uncertainty	Unit 2 Uncertainty
1	Hydraulics: Profile factor	0.25%	0.20%	0.19%
2	Geometry: Spool dimensions Spool piece alignment Spool piece thermal expansion	0.09%	0.18%	0.18%
3	Time Measurements Time of Flight Measurements Non-fluid delay	0.05%	0.11%	0.11%
4	Feedwater Density[2] [4] Feedwater Density/Correlation Feedwater Density/Temperature Feedwater Density/Pressure	0.07%	0.07%	0.07%
5	Subtotal: Mass Flow Uncertainty (Root Sum Square of Items 1-4)	0.28%	0.30%	0.29%
6	Feedwater Enthalpy[3] [4] Feedwater Enthalpy/Temperature Feedwater Enthalpy/Pressure Power Uncertainty, Thermal Expansion	0.08% 0.12%	0.09% 0.12%	0.09% 0.12%
7	Steam Enthalpy: Pressure input and moisture uncertainty[5]	0.07%	0.08%	0.08%
8	Gains/Losses	0.07%	0.07%	0.07%
9	Total Thermal Power Uncertainty	0.33%	0.35%	0.35%

1. Items 1 through 6 are directly associated with the Caldon LEFM CheckPlus System device. Items 7 and 8 are based on other plant process inputs.
2. Density errors due to the density correlation, the LEFM feedwater temperature determination and the feedwater pressure measurement.
3. Enthalpy errors due to the enthalpy correlation, the LEFM feedwater temperature determination and the feedwater pressure measurement.
4. The bounding uncertainties in pressure and temperature are +15 psi and +0.57°F, respectively.

Table I-1 (Continued)
Total Thermal Power Uncertainty Determination for Surry Units 1 and 2

Item	Parameter[(1)]	ER-157P, Rev. 5 Uncertainty	Unit 1 Uncertainty	Unit 2 Uncertainty
5. A steam moisture carryover uncertainty of 0.08% is assumed. Steam enthalpy uncertainty of 0.08% is the result of the square root of the sum of the squares of steam moisture uncertainty of 0.042% rated power and steam pressure uncertainty of 0.064% rated power.				

I.1.F Calibration and Maintenance Procedures of Instruments Affecting the Power Calorimetric

Information to specifically address the following aspects of the calibration and maintenance procedures related to the instruments that affect the power calorimetric.

I.1.F.i Maintaining Calibration

Calibration and maintenance for the UFM hardware and instrumentation will be performed using procedures based on the appropriate Cameron LEFM CheckPlus technical manuals, which ensures that the UFM remains bounded by the Topical Report ER-80P analysis and assumptions. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work is planned and executed in accordance with established SPS work control processes and procedures. Routine preventive maintenance activities for the UFM will include, but not be limited to, those activities specified in Section I.1.D.1.1.

I.1.F.ii Controlling Software and Hardware Configuration

The LEFM CheckPlus system is designed and manufactured per Cameron's 10 CFR 50, Appendix B, Quality Assurance Program and Verification and Validation (V&V) Program. Cameron's V&V Program fulfills the requirements of ANSI/IEEE-ANS Standard 7-4.3.2, 1993 (Reference I-16) and ASME NQA-2a-1990 (Reference I-17). After installation, the UFM software configuration will be maintained using existing procedures and processes, which include verification and validation of software configuration changes. UFM hardware and the calorimetric process instrumentation will be maintained per the SPS configuration control processes.

I.1.F.iii Performing Corrective Actions

Plant instrumentation that affects the power calorimetric, including the UFM inputs, will be monitored by SPS personnel. Problems detected are documented per the SPS corrective action process and necessary actions are planned and implemented.

I.1.F.iv Reporting Deficiencies to the Manufacturer

Conditions found to be adverse to quality will be documented per the SPS corrective action program and reported to the vendor as needed to support corrective action.

I.1.F.v Receiving and Addressing Manufacturer Deficiency Reports

SPS has existing processes for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the SPS corrective action program and actions will be controlled by the SPS work control process.

I.1.G Completion Time and Technical Basis

For Surry, a completion time of 48 hours is proposed for operation at any power level in excess of the current licensed power level of 2546 MWt with the UFM non-functional, provided that steady-state conditions persist throughout the 48-hour period selected. The basis for the proposed completion time of 48 hours follows.

- Operations procedures will direct the use of the back-up calorimetric in the event of UFM failure. This algorithm receives input from alternate plant instruments (feedwater venturis and RTDs) for feedwater flow rate calculation. The total feedwater flow from the three venturis will be normalized to the UFM feedwater flow rate, so that the alternate calorimetric matches the primary UFM based calorimetric. Also, the feedwater temperature and feedwater pressure measurements will be normalized to the more accurate data from the UFM. Alternate instrumentation accuracy due to nozzle fouling or transmitter drift will not result in significant calorimetric measurement uncertainty over a 48-hour period. The North Anna feedwater flow venturis were inspected in fall 2007 (Unit 1) and fall 2008 (Unit 2) during recalibration, which was the first visual inspection since plant startup. No venturi fouling was identified. There have been no noticeable changes in calorimetric power at Surry in the past several years, which leads to the conclusion that no significant fouling has occurred at Surry Units 1 or 2 either. Based on the recent North Anna inspection results and the consistent calorimetric power at Surry, it is very unlikely that venturi fouling or defouling would occur during the 48-hour completion time. A feedwater flow transmitter drift study showed the average drift was 0.017% and 0.014% respectively, for the first and second cycles used in the study. The maximum observed value for a single transmitter over one cycle was 0.05%. If all 6 feed flow transmitters were biased by the maximum observed value of 0.05%, the effect on calculated power would be essentially the same, i.e., a 0.05% change in power over the 18-month fuel cycle. This leads to the conclusion that transmitter drift over a 48-hour period would be negligible.

- Most UFM repairs are expected to be completed within an 8-hour shift. A completion time of 48 hours provides plant personnel sufficient time to plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.
- The normalized calorimetric instrumentation retains the accuracy of the UFM above 90% RP. However, if the plant experiences a power decrease below 2546 MWt (98.4% of the uprated RP) during the 48-hour period, the maximum permitted power level will be the current licensed core power level of 2546 MWt. This simplifies the Technical Requirements Manual statements for Applicability, Condition, Required Action and Completion Time. Further, it is conservative to limit the power level to ≤98.4% RP until the UFM is returned to functional status.
- As described in Cameron Report ER-157P (Reference I-2), the UFM consists of two planes of transducers. Although a single path or single plane malfunction results in a minimal increase in feedwater flow uncertainty, operators will conservatively respond to a failure of a single path, single plane, or single spool piece in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode.

The Technical Requirements Manual (TRM) uses the term "functional" for a system, structure, or component (SSC) that is not controlled by Technical Specifications. An SSC is functional when it is capable of performing its specified function, as set forth in the current licensing basis. TRM 3.3.5 will provide the plant administrative controls for the Feedwater UFM Calorimetric. Consistent with TRM 3.3.5, the Feedwater UFM Calorimetric shall be functional with: (a) the Feedwater UFM system functional; and (b) the plant computer system (PCS) calorimetric program functional. Thus, a failure of either the UFM system or the PCS calorimetric program will result in the Feedwater UFM Calorimetric being declared "not functional." The following excerpt is the proposed Basis for TRM 3.3.5 and describes the conditions that exist for a non-functional UFM.

> The Feedwater UFM System performs on-line self diagnostics to verify system operation within design basis uncertainty limits. Any out of specification condition will result in a control room annunciator. A failure between the Feedwater UFM System electronics cabinet and the plant computer will also result in a control room annunciator. If the feedwater UFM failure annunciator is received, the Feedwater UFM System will be

declared not functional. The control room annunciator response procedure provides guidance to the operators for initial alarm diagnosis and response.

Although a single plane malfunction results in a minimal increase in feedwater flow uncertainty, operators will conservatively respond to a single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode.

The 48-hour completion time will begin at the time the failure is annunciated in the main control room. A control room annunciator response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine UFM status and cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operator and maintenance response actions.

I.1.H Actions for Exceeding Completion Time and Technical Basis

The UFM functional requirements will be contained in the SPS Technical Requirements Manual (TRM). The LCO for the TRM states that an operable UFM shall be used to perform the daily calorimetric heat balance measurements. If the UFM is declared non-functional, the LCO will require that either the UFM is restored to functional status within 48 hours or power is reduced to ≤2546 MWt, which is the current licensed core power level and is 98.4% of the uprated RP of 2587 MWt. Operation at ≤2546 MWt is required to ensure that the plant safety analysis is bounding with a power calorimetric uncertainty that is supported by the feedwater flow or main steam flow venturis and temperature measurements.

In the event the UFM is not functional, the inputs to the power calorimetric will be determined by alternate instrumentation. The existing feedwater venturi flow nozzles and RTDs or the main steam flow venturi will be used for the calorimetric until the UFM is returned to functional status. Surry Units 1 and 2 have the option to use either steam or feed flow as input to the calorimetric power calculation. Within the first 48 hours after the identification of a non functional UFM, normalized main feed flow will be used. After the 48-hour completion time expires, Surry has the option to base the calorimetric calculation off of steam or feed flow input. To ensure that the feedwater venturi based calorimetric is consistent with the UFM based calorimetric, the feedwater venturi based flow rate, feedwater temperature, and feedwater pressure will be normalized to the UFM. A plant computer loss is treated as a loss of both the UFM and the ability to obtain corrected calorimetric power using the alternate plant instrumentation. Operation at the uprated power level may continue until the next required nuclear instrumentation heat balance which occurs daily. The plant computer failure will require reducing core thermal power to less than or equal to 2546 MWt as needed to support a manual calorimetric power calculation. These requirements ensure that a functional low uncertainty input is used whenever core power is greater than the current licensed core power level of 2546 MWt. The reactor operators will be provided with procedural guidance for those occasions when the UFM is not functional.

I REFERENCES

I-1 Cameron Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System*, Caldon Inc., March 1997.

I-2 Cameron Engineering Report ER-157P, Revision 5, *Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System*, Caldon Inc., October 2001.

I-3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C. L. Terry, TU Electric, Comanche Peak Steam Electric Station, Units 1 and 2 – *Review of Caldon Engineering Topical Report ER 80P, Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System* (ADAMS accession number 9903190065, legacy library), March 8, 1999.

I-4 Letter from S. A. Richards, NRC, to M. A. Krupa, *Entergy, Waterford Steam Electric Station, Unit 3; Riverbend Station; and Grand Gulf Nuclear Station – Review of Caldon Engineering Report ER-157P* (ML013540256), December 20, 2001.

I-5 NRC Letter, B. E. Thomas to E. M. Hauser, *Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus Ultrasonic Flow Meters (UFM)* (ML061700222), July 5, 2006.

I-6 Cameron Engineering Report ER-650, Revision 2, *Bounding Uncertainty Analysis for Thermal Power Determination at Surry Unit 1 Using the LEFM CheckPlus System*, September 2009.

I-7 Cameron Engineering Report ER-651, Revision 1, *Bounding Uncertainty Analysis for Thermal Power Determination at Surry Unit 2 Using the LEFM CheckPlus System*, September 2009.

I-8 Cameron Engineering Report ER-684, Revision 2, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Surry Unit 1*, September 2009.

I-9 Cameron Engineering Report ER-690, Revision 2, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Surry Unit 2*, September 2009.

I-10 *Cameron Customer Information Bulletin CIB-125*, Revision 0, April 23, 2007.

I-11 NRC letter to FPL Energy Seabrook, *Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML061360034, May 22, 2006.

I-12 NRC letter to Vogtle Electric Generating Plant, Units 1 and 2, *Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate*, ML080350347, February 27, 2008.

I-13 NRC letter to Nebraska Public Power District, Cooper Nuclear Station, *Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate*, ML081540280, June 30, 2008.

I-14 ISA-RP67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation.*

I-15 Dominion Technical Report EE-0116, Revision 4, *Allowable Values for North Anna Improved Technical Specifications (ITS) Tables 3.3.1-1 and 3.3.2-1 and Setting Limits for Surry Custom Technical Specifications (CTS)*, Sections 2.3 and 3.7, September 30, 2008.

I-16 ANSI/IEEE-ANS Standard 7-4.3.2, 1993, *IEEE Standard Criteria for Digital Computers in Safety System of Nuclear Power Generating Station.*

I-17 ASME NQA-2a-1990, *Quality Assurance Requirement for Nuclear Facility Application.*

II ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

   A. Identify the transient or accident that is the subject of the analysis

   B. Confirm and explicitly state that

      i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

      ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

      iii. the analyses of record are not changed by the requested power uprate

   C. Confirm that bounding event determinations continue to be valid

   D. Provide a reference to the NRC's previous approvals discussed in Item B above

RESPONSE TO II - ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

II. Accidents and Transients Bounded by the Analyses of Record for the Measurement Uncertainty Recapture

II.1 Introduction

A review of UFSAR Chapters 5, 6, and 14 and other related subsections was performed to support the SPS MUR power uprate with respect to the accident analyses. Evaluations were also performed on other analyses (e.g., internal flooding, SBO, ATWS). The UFSAR review was conducted to confirm that the existing analyses of record, as currently presented in the UFSAR, were performed conservatively and remain valid and bounding for the proposed power uprate. Table II-1 indicates the analysis power levels used for the SPS MUR power uprate.

Table II-1
Analysis Power Levels for Surry Units 1 and 2 MUR Uprating

Analysis Scope	Core Power MWt	NSSS Power MWt(3)	Source
NSSS	2597(1)	2609	NSSS Design Parameters
Safety Analyses	2596.9(1)	2609	UFSAR Chapters 5, 6 and 14
Statistical DNBR Events	2589.3(2)	2602	UFSAR Chapter 14
Safety-Related System Evaluations	2597(1)	2609	Consistent with UFSAR safety analyses
BOP System Evaluations	2589.3(2)	2602	

1. 102% of current RP of 2546 MWt; while many safety analyses assume 2597 MWt, some safety analyses assume 2596.9 MWt, which is identified as the limiting analysis value in the table.
2. 101.7% of current RP of 2546 MWt
3. The analyses use 12 MWt for RCP net heat addition. For the BOP system evaluations, the NSSS power is rounded up to the next whole number

The analyses generally model the core and/or NSSS thermal power in one of three ways. First, some analyses apply a 2.0% increase to the initial power level to account for the power measurement uncertainty. These analyses have not been re-performed for the MUR uprate conditions, because the sum of the proposed core power level and the decreased power measurement uncertainty falls within the previously analyzed conditions. The existing 2.0% uncertainty is reallocated so a portion is applied to uprate power and the remainder is retained to accommodate the power measurement uncertainty. Second, some analyses employ a nominal power level. These analyses have either been evaluated or re-performed for the proposed power level. Third, some of the analyses are performed at 0% power conditions or do not actually model core power level. These analyses have not been re-performed because they are unaffected by the core power level.

For the SPS MUR power uprate, a core RP of 2587 MWt was selected based on the calorimetric uncertainty of 0.35% with the UFM and a review of the accident analysis assumptions for core power. The deterministic accident analyses use a minimum value of 2596.9 MWt (102% of 2546 MWt) as the total core power, which leaves 9.9 MWt of margin to accommodate the power uncertainty. The 9.9 MWt is 0.38% of 2587 MWt. Since the power calorimetric uncertainty of 0.35% at RP with the UFM is less than the accident analysis allowance of 0.38% with a 2587 MWt licensed power level, the deterministic accident analyses are bounding for the MUR power uprate. The statistical DNBR events were analyzed previously at 2546 MWt. Refer to Sections II.2.2, II.2.4, II.2.7, II.2.8, II.2.9, II.2.10, and II.2.11 for the treatment of statistical DNBR events at the proposed RP of 2587 MWt. In conclusion, the evaluations of the UFSAR events in Section II.2 support an uprated RP of 2587 MWt.

Table II-2 below provides a brief overview of the accident/transient analyses and other analyses contained in the SPS UFSAR (Reference II-1), the assumed core power level in each analysis, and whether these analyses remain bounding for the MUR power uprate. This table also provides references to the NRC's previous approval of each analysis, if applicable, or a statement confirming that NRC approved methods were used in the analysis of record that was implemented under the provisions of 10 CFR 50.59. A discussion of each UFSAR event is presented in Section II.2, Discussion of Events.

Table II-2
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
Uncontrolled Control-Rod Assembly Withdrawal From a Subcritical Condition	14.2.1	0	Yes	NRC approval was documented in Reference II-8. Subsequent evaluations have been performed under the provisions of 10 CFR 50.59.
Uncontrolled Control-Rod Assembly Withdrawal at Power	14.2.2	102/100[1]	Yes/No[1]	NRC approval was documented in Reference II-8. Subsequent evaluations have been performed under the provisions of 10 CFR 50.59.
Malpositioning of the Part Length Control Rod Assemblies	14.2.3	N/A	N/A	The part length control rod assemblies have been removed from the core and no longer require any design evaluation.
Control Rod Assembly Drop/Misalignment	14.2.4	100[2]	No[2]	Dropped rod limit lines were approved by the NRC in Reference II-8. The dropped rod event is analyzed for each core reload using the NRC-approved methodology in Reference II-22.
Chemical and Volume Control System Malfunction	14.2.5	102	Yes	NRC approval was documented in Reference II-8.
SUIL Accident Analysis Design Basis	14.2.6	N/A	N/A	This event is only evaluated at cold shutdown and refueling shutdown conditions. Technical Specification 3.1.A.4.b (Reactor Coolant System) prohibits power operation with less than three reactor coolant loops in service.
Excessive Heat Removal Due to Feedwater System Malfunctions	14.2.7	102/100[3]	Yes/No[3]	Analyses were performed using NRC-approved methodologies.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
Excessive Load Increase Incident	14.2.8	100[2]	No[2]	Analysis was performed using NRC-approved methodologies.
Loss of Reactor Coolant Flow	14.2.9	102/100[1]	Yes/No[1]	NRC approval was documented in Reference II-8. Subsequent evaluations have been performed under the provisions of 10 CFR 50.59.
Environmental Consequences of Locked Rotor Accident (LRA)	14.2.9.2.4	102.3	Yes	NRC approval in Reference II-8.[12]
Loss of External Electrical Load	14.2.10	102/100[1]	Yes/No[1]	Analysis was performed using NRC-approved methodologies.
Loss of Normal Feedwater	14.2.11	102	Yes	Analysis was performed using NRC-approved methodologies.
Loss of All Alternating Current Power to the Station Auxiliaries	14.2.12	102	Yes	Analysis was performed using NRC-approved methodologies.
Likelihood of a Turbine-Generator Unit Overspeed	14.2.13	N/A	N/A	The turbine-generator speed is constant and not dependent upon reactor power level.
Steam Generator Tube Rupture	14.3.1	102	Yes	Analysis was performed using NRC-approved methodologies.
Environmental Consequences of Steam Generator Tube Rupture (SGTR)	14.3.1.4	102[13]	No	NRC review required.
Rupture of Main Steam Pipe	14.3.2	0	Yes	Analysis was performed using NRC-approved methodologies.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
Environmental Consequences of a Main Steam-Line Break (MSLB)	14.3.2.4	102[13]	No	NRC review required.
Rupture of a Control Rod Drive Mechanism Housing (Control Rod Assembly Ejection)	14.3.3	102	Yes	Analysis was performed using NRC-approved methodologies.
Fuel Handling Accidents	14.4.1	102.3	Yes	NRC approval in Reference II-27.
Volume Control Tank Rupture	14.4.2.1	NA[10]	Yes	NRC approval in Reference II-29.[11]
WGDT Rupture	14.4.2.2	NA[8]	Yes	NRC approval in Reference II-29.[9]
Radioactive Liquid Release	14.4.3	N/A	N/A	This UFSAR section merely describes the various plant design features used to prevent a liquid release impacting offsite dose levels. There is no analysis associated with this item.
Large Break LOCA (long-term cooling)	14.5	102	Yes	Analyses for post-LOCA containment sump boron concentration, containment sump pH, and hot leg switchover time were approved by the NRC in Reference II-9. The analyses have been supplemented by additional evaluations under the provisions of 10 CFR 50.59.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
Major Reactor Coolant System Pipe Ruptures (Large Break Loss-of-Coolant Accident)	14.5.1	102	Yes	The large break loss-of-coolant accident PCT analysis of record was performed with the NRC approved methodology in Reference II-4, which was approved by the NRC for use at Surry in Reference II-5. The analysis has been supplemented by additional evaluations under the provisions of 10 CFR 50.46.
Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes, Which Actuates Emergency Core Cooling System (SBLOCA)	14.5.2	102	Yes	The SBLOCA PCT analysis was performed with the NRC approved methodology in References II-6 and II-7, which was approved by the NRC for use at Surry in Reference II-8. The analysis has been supplemented by additional evaluations under the provisions of 10 CFR 50.46.
Environmental Consequences of LOCA	14.5.5	102.3	Yes	NRC approval in Reference II-3.
Natural Circulation Cooldown	14.2.12	102	Yes	Analysis was performed using NRC-approved methodologies.
Long-term LOCA Mass and Energy Release	5.4.1	102	Yes	NRC approval in Reference II-3 and supplemented by reanalysis implemented under the provisions of 10 CFR 50.59.
Short-term LOCA Mass and Energy Release	15.6.3	(4)	Yes	NRC approval in Reference II-8.
Main Steam Line Break Mass and Energy Release	5.4.3	116.1	Yes	NRC approval in Reference II-3.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
ATWS/AMSAC	7.2.3.2.5 7.2.3.2.7	109.4	Yes	NRC approval in Reference II-12.
SBO	8.4.6	102	Yes	The 8-hour SBO analysis for the minimum ECST volume was described in Reference II-30 and reviewed by the NRC in Section 2.3 of Reference II-31. The analysis has been supplemented by additional evaluations under the provisions of 10 CFR 50.59.
Analyses to Determine EQ Parameters	7.5.3.5	100	No	NRC approval in Reference II-36.
Reactor Coolant Activity Concentration, Monitoring, and Control	9.1.2.2	100	No	NRC review required.
Safe Shutdown Analysis (Appendix R Fire report)	9.10(5)	100	No(6)	The NRC Safety Evaluation Reports for 10 CFR 50 Appendix R compliance are documented in References II-23, II-24, and II-25.
Spent Fuel Pool Cooling	9.5	102	Yes	NRC approval was documented in Reference II-29. Subsequent evaluations have been performed under the provisions of 10 CFR 50.59.
Internal Flooding	2.3.1.2	N/A	Yes	NRC approval in Reference II-37.
HELB in the MSVH	14B.6	102	Yes	Analysis was performed using NRC-approved methodologies.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval
Rupture of a Feedwater Pipe	N/A	102	Yes	NRC approval was documented for the core uprate in Reference II-8.

1. Analyses for RCS and MSS overpressure were performed at 102% of 2546 MWt and are bounding for the MUR power uprate. DNBR analyses were performed at current RP of 2546 MWt, consistent with the statistical DNBR analysis methodology (Reference II-15). The UFSAR analyses of record for DNBR do not need to be revised. A retained DNBR margin penalty for a maximum MUR power uprate of 1.7%, or 2589.3 MWt, is applied to these statistically-treated DNBR events.
2. These events are analyzed for DNBR only at current RP of 2546 MWt, consistent with the statistical DNBR analysis methodology (Reference II-15). The UFSAR analyses of record do not need to be revised. A retained DNBR margin penalty for a maximum MUR power uprate of 1.7%, or 2589.3 MWt, is applied to these events.
3. The excessive feedwater flow event was analyzed at 102% of 2546 MWt, consistent with the methodology for deterministic events. The feedwater temperature reduction event was analyzed at the current RP of 2546 MWt, consistent with the statistical DNBR evaluation methodology (Reference II-15). The analysis of record for a feedwater temperature reduction event is not revised. A retained DNBR margin penalty for a maximum MUR power uprate of 1.7%, or 2589.3 MWt, is applied to this event.
4. The short-term LOCA mass and energy releases are a function of RCS cold leg and hot leg temperatures which are a function of RCS flow and core power. Evaluations confirmed that the UFSAR analyses for short-term LOCA mass and energy releases and containment subcompartment response remain bounding for the RCS design temperatures for the MUR power uprate.
5. The Appendix R safe shutdown analyses are not described in UFSAR Section 9.10, where the fire protection equipment is specified. The post-fire safe shutdown analyses are maintained in engineering calculations.
6. The safe shutdown analyses that support the Appendix R program were reviewed. The review concluded that the results of the safe shutdown analyses remain valid for a bounding MUR uprated power of 2597 MWt.
7. The source term is limited by the 1 Ci/gm Dose Equivalent I-131 TS limit and the steam release is modeled at 102% of current rated thermal power.

Table II-2 (Continued)
UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (% of 2546 MWt)	Bounding (Yes/No)	NRC Approval

8. Not dependent upon core power; based on TS 3.11.B, which limits the Gas Storage Tanks to 24,600 curies of noble gases (considered as Xe-133).
9. The original WGDT rupture analysis was submitted in the FSAR, which was approved by NRC in Reference II-29. The WGDT rupture analysis was subsequently updated under the provisions of 10 CFR 50.59.
10. Based on 1% failed fuel fission product inventory in the RCS. See Section II.2.24.6 for further discussion.
11. The original VCT rupture analysis was submitted in the FSAR, which was approved by NRC in Reference II-29. However, the NRC did not specifically discuss the VCT rupture analysis in Reference II-29. The VCT rupture analysis was subsequently updated under the provisions of 10 CFR 50.59.
12. The original Locked Rotor Accident (LRA) was approved by NRC in Reference II-8. The LRA analysis has been subsequently updated to AST methods under the provisions of 10 CFR 50.59.
13. The source term is limited by the 1 μCi/gm Dose Equivalent I-131 TS limit and the steam release is modeled at 102% of current rated thermal power.

II.2 Discussion of Events

UFSAR Chapter 14 accidents were reviewed to support the SPS power uprate. A summary of each accident evaluation is provided below.

Summary of DNBR Analysis Basis for Surry Improved Fuel (SIF)

Table II-3 identifies the Surry UFSAR Chapter 14 transients that are analyzed to verify that departure from nucleate boiling (DNB) ratio (DNBR) limits are met, the DNB methodology that is applicable (statistical or non-statistical/deterministic), and the DNB correlation used in the analysis of record. DNBR analyses for the Westinghouse 15x15 Surry Improved Fuel (SIF) product use the NRC-approved COBRA code and the W-3 (Reference II-32) or WRB-1 (Reference II-33) correlation, depending on the transient. The Virginia Power Statistical DNBR Evaluation Methodology in topical report VEP-NE-2-A (Reference II-34) is applied to all statistically-treated events. The NRC approved the use of this methodology and the current DNBR limits for Surry in Reference II-35.

Table II-3
Surry UFSAR Transients Analyzed for DNBR

UFSAR Section	Event Description	Stat/Non-Stat DNB Methods	COBRA Model	DNB Correlation
14.2.1	Rod Withdrawal from Subcritical (RWSC)	Non-Stat	SIF	W-3
14.2.2	Rod Withdrawal at Power (RWAP)	Stat	SIF	WRB-1
14.2.4	Dropped Rod/Misaligned Rod	Stat	SIF	WRB-1
14.2.7	Excessive Heat Removal:			
	• Full Power Excessive Flow	Non-Stat	SIF	WRB-1
	• Full Power Temperature Reduction	Stat	SIF	WRB-1
14.2.8	Excessive Load Increase	Stat	SIF	WRB-1
14.2.9.1	Loss of Flow (LOFA)	Stat	SIF	WRB-1
14.2.9.2	Locked Rotor (LRA)	Stat	SIF	WRB-1
14.2.10	Loss of Load (LOL)	Stat	SIF	WRB-1
14.3.2	Main Steamline Break (MSLB)	Non-Stat	SIF	W-3
Tech Specs	Core Thermal Limits	Stat	SIF	WRB-1
Tech Specs	OTΔT and OPΔT setpoints and f(ΔI) function	Stat	SIF	WRB-1

Method for Evaluating the MUR Power Uprate Effect on DNBR

Topical Report VEP-NE-2-A (Reference II-34) describes the calculation of "retained DNBR margin" as the difference between the DNBR design limit and the statistical design limit. The available retained DNBR margin is evaluated for each reload core, considering DNBR penalties for generic fuel design issues (e.g., fuel rod bow), cycle-specific violations of limits (e.g., fuel rod power census), and plant-operating conditions. Surry UFSAR Section 3.4.3.5 also summarizes the applicable uses of retained DNBR margin.

The statistical DNBR design limit is 1.27 for the 15x15 SIF product with the COBRA/WRB-1 code/correlation set. The DNBR design limit is 1.46. These limits are described in Surry UFSAR Sections 3.2.3.3 and 3.4.3.5 and were approved by the NRC in Reference II-35. The difference between these DNBR limits is 13.0% retained DNBR margin that is used in the core reload thermal-hydraulic evaluation in accordance with Reference II-34.

The approach for the MUR power uprate is to develop a penalty against retained DNBR margin. With the use of retained DNBR margin to accommodate the power uprate, the UFSAR Chapter 14 DNBR analyses of record are not affected. Based on a review of NRC Regulatory Issue Summary 2002-03, Dominion concludes that the DNBR penalty for the MUR power uprate should be described in Section II of the License Amendment Request.

<u>Effect of MUR Power Uprate on DNBR</u>

The method involves calculating the bounding effect on DNBR for a 1.7% core power increase above 2546 MWt (current RP). The change in DNBR from a change in core power can be quantified using Equation 1.

$$\Delta DNBR = \frac{\partial(DNBR)}{\partial(Power)} \times \Delta Power \qquad \text{Equation 1}$$

Note that ΔDNBR and ΔPower are in percent. The DNBR-Power partial [∂(DNBR)/∂(Power)] quantifies the percent change in DNBR as the result of a percent change in power with other parameters constant. The calculation of the DNBR-Power partial is described by Equation 2.

$$\frac{\partial(DNBR)}{\partial(Power)} = \frac{DNBR_2 - DNBR_1}{Power_2 - Power_1} \times \frac{Power_1}{DNBR_1} \qquad \text{Equation 2}$$

$DNBR_1$ is the DNBR calculated at $Power_1$ and $DNBR_2$ is the DNBR calculated at $Power_2$. $DNBR_1$ is calculated with a COBRA analysis with Power = $Power_1$ at a specific statepoint. $DNBR_2$ is calculated with COBRA by perturbing power 1% ($Power_2 = Power_1 * 1.01$) at the same statepoint conditions. The DNBR-Power partial then is calculated using Equation 2.

The DNBR-Power partials are calculated at the following statepoints that represent normal operation and limiting accident conditions:

- Nominal design, hot full power conditions;
- The minimum DNBR (MDNBR) statepoints for the loss of flow accident (LOFA) from UFSAR Section 14.2.9.1;
- The MDNBR statepoint for the rod withdrawal at power (RWAP) event from UFSAR Section 14.2.2; and
- Statepoints on the DNBR limit lines from 118% to 90% power and some vessel exit boiling points along the core thermal limit lines.

DNBR analyses used the NRC-approved COBRA methodology (References II-32 and II-33) that is the existing licensing basis for UFSAR Chapter 14 analyses of 15x15 SIF. The MDNBR results of the current COBRA analyses are used as $DNBR_1$ to calculate the DNBR-Power partial with Equation 2. From the 19 statepoints analyzed, the maximum DNBR-Power partial was calculated as -1.92% DNBR/% power along the 118% DNBR limit line. The maximum DNBR-Power partial is used to determine a penalty against retained DNBR margin. Using Equation 1, the maximum DNBR-Power partial is used to determine the effect on DNBR for a bounding 1.7% power uprate.

$$\Delta DNBR = \frac{\partial(DNBR)}{\partial(Power)} \times \Delta Power = -1.92\%/\% \times 1.7\% = -3.27\%$$

A DNBR penalty of 3.3% will be applied for a bounding 1.7% power uprate above 2546 MWt. This DNBR penalty is conservative and applicable to all statistically-treated DNBR events and will be deducted from the retained DNBR margin during the core reload thermal-hydraulic evaluation in accordance with the NRC-approved methodology in VEP-NE-2-A (Reference II-34). It has been confirmed that Surry has adequate retained DNBR margin to accommodate the MUR penalty.

Review of Power Uncertainty in DNBR Design Limit

Section 3.4.3.2 of the Surry UFSAR identifies a 2% power calorimetric uncertainty that is used for transients that are analyzed with deterministic DNBR uncertainties. Deterministic DNBR events analyzed at full power will continue to be based on 102% of 2546 MWt. The statistical DNBR limit for the COBRA/WRB-1 code/correlation set was developed consistent with the Virginia Power Statistical DNBR Evaluation Methodology (Reference II-34) and used an uncertainty of ±2.4% at 2σ standard deviation with respect to 2546 MWt (current RP). Although the power calorimetric uncertainty will decrease with the use of feedwater ultrasonic flow meters, the statistical DNBR limit for COBRA/WRB-1 will not change and continue to be based on ±2.4% at 2σ standard deviation for conservatism. The NRC approved the DNBR limits based on this uncertainty in Reference II-35.

II.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition - UFSAR 14.2.1

The RWSC event is analyzed for comparison to DNBR and RCS pressure limits. The system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17). The DNBR analysis was performed using the NRC-approved COBRA methodology with the W-3 correlation (Reference II-13). The analyses were described in Section 3.5.1 of Attachment 3 of the license amendment for the core power uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-8. Because this event is evaluated at HZP conditions (0% rated core power), the UFSAR analysis of record for the RWSC event is unaffected by the MUR power uprate.

II.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power - UFSAR 14.2.2

The Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (RWAP) event is analyzed for comparison to DNBR, RCS pressure, and MSS pressure limits. The system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17). The DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) with the WRB-1 correlation (Reference II-14). The analyses were described in Section 3.5.2 of Attachment 3 of the license amendment for the core uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-8.

RCS and MSS overpressure cases were performed at 10%, 12%, 60%, 100%, and 102% of 2546 MWt core power. The limiting case for RCS overpressure occurs at 12% of 2546 MWt. The limiting cases for MSS overpressure occur at 10% and 60% of 2546 MWt. The analyses for RCS and MSS overpressure demonstrate that the reactor protection system provides adequate protection and the analyses remain bounding for the MUR power uprate.

DNBR analyses were performed at 10%, 60% and 100% of 2546 MWt core power. The use of the nominal full power value is consistent with the NRC-approved statistical DNBR methodology in Topical Report VEP-NE-2-A (Reference II-15). As discussed in Section VIII, a change to the Overtemperature ΔT (OTΔT) trip pressure constant (K3) is required to ensure protection at low RCS pressures for the MUR power uprate and plant operating conditions. To evaluate the effect of the OTΔT K3 pressure constant change, Dominion reanalyzed the RWAP event to confirm adequate protection for a range of thermal hydraulic conditions. Because explicit analysis of the RWAP event at 101.7% of 2546 MWt demonstrated minimum DNBR remains above the DNBR design limit, a penalty against retained DNBR margin is not required for the RWAP event. Analysis details are provided in Section VIII.

II.2.3 Malpositioning of the Part Length Control Rod Assemblies - UFSAR 14.2.3

This event was originally evaluated in SPS UFSAR Section 14.2.3. The part length control rod assemblies have since been removed from the core and no longer require any design evaluation.

II.2.4 Control Rod Assembly Drop/Misalignment - UFSAR 14.2.4

Control-rod misalignment accidents include (1) dropped full length assemblies, (2) dropped full-length assembly groups, and (3) statically misaligned assemblies.

Dropped Rod

The dropped rod cluster control assembly (RCCA) event is conservatively evaluated with three analyses–transient, nuclear, and thermal/ hydraulic–that provide: (1) statepoints (reactor power, pressure, and temperature) at the most limiting time in the transient; (2) the radial peaking factor at the most limiting conditions in the transient; and, (3) the DNB analysis at the conditions determined by Steps 1 and 2. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference II-22.

A single or multiple dropped rod/RCCA is evaluated through the development of dropped rod limit lines at full power. For a particular plant, any combination of $F\Delta H$, inlet temperature, and pressure can be used to determine a fractional core power level for which the DNBR is equal to the DNBR limit for a specified reference power shape. These results are plotted as inlet temperature versus power level at constant pressure and $F\Delta H$. These curves, otherwise known as "dropped rod limit lines," are then used to determine the allowable radial peaking factor at the limiting point during the dropped rod transient. For each core design, an evaluation is performed to demonstrate that the reference power shape that was used to generate the dropped rod limit lines remains bounding relative to the reload power shapes. Then, a rod drop analysis is performed with cycle-specific inputs to demonstrate that the limiting pre-drop $F\Delta H$ values are above predicted reload values for the fuel cycle.

The dropped rod limit lines in the current analysis are based on a rated core power of 2546 MWt. Use of nominal power is consistent with the NRC-approved statistical DNBR analysis methodology in topical report VEP-NE-2-A (Reference II-15). To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for the dropped rod event, and the current dropped rod limit lines do not need to be revised.

Misaligned Rod

A misaligned rod results in skewed axial and radial power profiles and causes power peaking in certain regions of the core. To assure that the worst case misaligned rod position does not exceed the design limits of the core, an enthalpy factor $F\Delta H$, corresponding to this worst case misalignment, is calculated for each cycle. The peak radial power factor, $F\Delta H$, resulting from a rod out of position is calculated for each reload and is compared to the $F\Delta H$ limit that includes 4% calculational uncertainty. The 4% measurement uncertainty is included statistically by the Statistical DNBR Evaluation Methodology (Reference II-15). The $F\Delta H$ limit is based on a DNBR limit of 1.46 at nominal full power of 2546 MWt, which is consistent with the NRC-approved methodology in topical report VEP-NE-2-A (Reference II-15). To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for the misaligned rod event.

II.2.5 Chemical and Volume Control System Malfunction - UFSAR 14.2.5

UFSAR Section 14.2.5 considers a boron dilution during refueling, cold shutdown, intermediate shutdown, hot shutdown, reactor critical, and power operation. The boron dilution event is evaluated to verify the limits for DNBR and RCS and MSS overpressurization are met. During refueling and cold shutdown, the primary grade water flow path is locked out, procedurally preventing a boron dilution. In intermediate shutdown and hot shutdown conditions, the shutdown margin requirements ensure that at least 15 minutes are available from the identification of a dilution event to loss of shutdown margin. The adequacy of the administrative shutdown margin requirement is verified for each reload core and will continue to be verified for the MUR uprate cores. At reactor critical conditions, the consequences are bounded by the RWSC event in UFSAR Section 14.2.1, because the maximum achievable reactivity insertion rate experienced during a boron dilution event (<2.0 pcm/sec) is less than the reactivity insertion rate assumed in the RWSC analysis.

The "at power" boron dilution reactivity transient is essentially identical to that of a control rod assembly withdrawal (RWAP) that is analyzed in UFSAR Section 14.2.2. If the reactor is in manual control and the operator takes no action to correct an inadvertent boron dilution, the power and temperature will rise to the OTΔT trip setpoint. Because the maximum boron dilution reactivity insertion rate is within the range analyzed for the RWAP event (UFSAR Section 14.2.2), the DNBR, RCS pressure, and MSS pressure response would be bounded by the RWAP event and no explicit analysis of the boron dilution event is required. Therefore, it can be concluded that fuel cladding integrity is maintained during postulated boron dilution events in all operating modes, that RCS and MSS pressures remain below 110% of design pressure during postulated boron dilution events in all operating modes, and that the 15-minute operator response time is not affected by the MUR power uprate. The basis is consistent with the evaluation described in Section 3.5.4 of Attachment 3 of the license amendment for the core uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-8.

II.2.6 Start-up of an Inactive Reactor Coolant Loop - UFSAR 14.2.6

The current SUIL design and licensing bases credit Technical Specification controls to preclude the possibility of a significant inadvertent reactivity addition during or following loop stop valve operations. Technical Specification 3.1.A.4.b (Reactor Coolant System) prohibits operation with less than three reactor coolant loops in service. This prohibition is further discussed in UFSAR Section 14.2.6. The SPS power uprate review has not included any analyses that would provide the licensing basis for two loop operation. The initial assumptions for FSAR analyses are that the plant is maintained within the Technical Specification limits. Since two loop operation is and will remain prohibited by Technical Specifications, no analysis of this event is required for uprated conditions. Start-up of an inactive reactor coolant loop is a deliberate action under operator control governed by Technical Specifications, thus the sequence of operator errors required for a SUIL event to occur is considered non-credible. The start-up of an inactive reactor coolant loop is therefore not affected by the MUR power uprate.

II.2.7 Excessive Heat Removal Due to Feedwater System Malfunction - UFSAR 14.2.7

Reductions in feedwater temperature or additions of excessive feedwater can result in increases of core power above full power. Such transients are attenuated by the thermal capacity in the secondary plant and in the RCS. The overpower-overtemperature protection prevents any power increase that could lead to a DNBR less than the DNBR limit. These events are analyzed to confirm that the DNBR limits are met.

For the excessive feedwater flow analysis, the system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17) and the DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) and the WRB-1 correlation (Reference II-14). The full-power analyses were performed at 102% of 2546 MWt core power (consistent with deterministic DNBR analysis methods) with a minimum DNBR greater than the DNBR design limit. This accident analysis was performed before the implementation of statistical DNBR methods under Reference II-16; thus, the application of deterministic DNBR methods was appropriate. The excessive feedwater flow analysis is bounding for the MUR power uprate.

For the feedwater temperature reduction event, the system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17) and the DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) and the WRB-1 correlation (Reference II-14). The full-power analyses were performed at 2546 MWt core power (consistent with statistical DNBR analysis methods). The feedwater temperature reduction produces a primary system load increase that is less than the 10% increase above full power that is analyzed in UFSAR Section 14.2.8, Excessive Load Increase Incident. Because the feedwater temperature reduction event is bounded by other analyses, no explicit analysis of the feedwater temperature reduction event was performed for the MUR power uprate. To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for this event.

II.2.8 Excessive Load Increase Incident - UFSAR 14.2.8

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the SG load demand. The reactor control system is designed to accommodate a 10% step load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. This event is analyzed to confirm that the DNBR limit is met.

The system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17) and the DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) with the WRB-1 correlation (Reference II-14). The full-power analyses were performed at 2546 MWt core power, consistent with the NRC-approved statistical DNBR analysis methodology in Reference II-15. The minimum DNBR is above the design limit. No explicit analysis of the event was performed for the MUR power uprate. To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for this event.

II.2.9 Loss of Reactor Coolant Flow - Flow Coastdown Incidents - UFSAR 14.2.9.1

The LOFA is characterized by the loss of forced circulation in one or more RCS loops resulting from a mechanical or electrical failure. If the reactor is at power, the immediate effect is a rapid increase in coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. Various reactor trips (undervoltage or underfrequency on RCP power supply buses, low reactor coolant loop flow, RCP circuit breaker opening) provide the necessary protection against a LOFA.

The LOFA event system transient response was performed using the NRC-approved RETRAN methodology (Reference II-17). The DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) with the WRB-1 correlation (Reference II-14). The analyses were described in Section 3.5.6 of Attachment 3 of the license amendment for the core uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-8. The analysis was initialized at a reactor power of 2546 MWt consistent with the NRC-approved statistical DNBR analysis methodology in Reference II-15. The LOFA event was not reanalyzed for the uprate power level. To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for the LOFA event.

II.2.10 RCP Locked Rotor Incident - UFSAR 14.2.9.2

The LRA events are characterized by the rapid loss of forced circulation in one RCS loop. A locked rotor event is defined as the seizure of a RCP motor due to a mechanical failure. The sheared shaft event is defined as the separation of the RCP impeller from the motor due to the severance of the impeller shaft. For both the locked rotor and the sheared shaft events, the postulated RCP failure causes the reactor coolant flow rate to decrease more rapidly than a normal RCP coastdown.

During power operation the reduction in RCS flow caused by a locked rotor or sheared shaft event results in degradation of the heat transfer between the fuel and the reactor coolant, and between the reactor coolant and the secondary coolant in the SG. As a result of the reduced fluid velocity, the core differential temperature (ΔT) and RCS average temperature (T_{avg}) increase. The reduced heat transfer to the secondary fluid also contributes to the reactor coolant temperature increase. The expansion of the RCS fluid that accompanies the temperature increase causes an in-surge of coolant into the pressurizer, and thus an increase in the RCS pressure. The reduced fluid velocity and subsequent temperature rise also act to reduce the heat transfer from the fuel, causing the fuel temperature to increase. Fuel damage could then result if specified acceptable fuel damage limits are exceeded during the transient, i.e., if the fuel experiences a DNB. Due to the severe nature of these postulated failures, the likelihood that a limited number of fuel rods will experience DNB is significant. Thus, timely actuation of the reactor protection system is required to help limit the number of potential fuel failures.

The system transient analysis for the LRA event was performed using the NRC-approved RETRAN methodology (Reference II-17). The DNBR analysis was performed using the NRC-approved COBRA methodology (Reference II-13) with the WRB-1 correlation (Reference II-14). The analyses were described in Section 3.5.7 of Attachment 3 of the license amendment for the core uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-8.

The RCS and MSS overpressure cases were analyzed with an initial core power of 102% of 2546 MWt, demonstrating margin to the RCS and MSS system pressure limits (i.e., 110% of design pressure). Thus, these cases are bounding for the MUR power uprate.

The DNBR analysis was initialized at a reactor power of 2546 MWt consistent with the NRC-approved statistical DNBR analysis methodology in Reference II-15. The LRA event was not reanalyzed for the uprate power level. To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for the LRA event.

II.2.11 Loss of External Electrical Load - UFSAR 14.2.10

Major load loss on the plant can result from loss of external electrical load (LOL) or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the RCPs. The LOL event is analyzed to confirm that DNBR, RCS pressure, and MSS pressure have margin to the applicable limits. The analysis of record used the NRC-approved RETRAN methodology (Reference II-17) for the plant transient response and the NRC-approved COBRA methodology (Reference II-13) with the WRB-1 correlation (Reference II-14) for the DNBR analysis. The MSS and RCS overpressure cases used an initial reactor power of 102% of 2546 MWt. Therefore, the RCS overpressure and MSS overpressure cases for the loss of load event are bounding for the MUR power uprate.

The DNBR analysis used an initial reactor power of 2546 MWt, consistent with the statistical DNBR evaluation methodology in Reference II-15. The LOL event was not reanalyzed for the MUR power uprate. To accommodate a maximum 1.7% power uprate, a DNBR penalty of 3.3% will be applied against retained DNBR margin for the LOL event.

II.2.12 Loss of Normal Feedwater - UFSAR 14.2.11

The loss of normal feedwater event is analyzed for RCS overpressure and pressurizer overfill leading to a loss of reactor coolant. The analysis of record used the NRC-approved RETRAN analysis methodology (Reference II-17) to determine the plant transient response. The analysis assumed an initial core power of 102% of 2546 MWt. Therefore, the loss of normal feedwater analysis is bounding for the MUR power uprate.

II.2.13 Loss of All Alternating Current Power to the Station Auxiliaries - UFSAR 14.2.12

In the event of a complete LOOP and a turbine trip, there will be a loss of power to the plant auxiliaries (the RCPs, condensate pumps, etc.). The events following a loss of AC power with turbine and reactor trip are described in Section 14.2.12 of the UFSAR. The main difference between this event and the loss of normal feedwater event in UFSAR Section 14.2.11 is that the loss of power trips the RCPs, which reduces the primary system heat load but requires natural circulation cooling. The loss of all AC power case is bounded by the loss of normal feedwater analysis in UFSAR Section 14.2.11, which was analyzed at 102% of 2546 MWt core power. The analysis is bounding for the MUR power uprate.

II.2.14 Likelihood of Turbine-Generator Unit Overspeed - UFSAR 14.2.13

SPS UFSAR Section 14.2.13 contains the assessment of this event. The existing analysis has accounted for the effects of turbine missiles generated at speeds up to 120% of rated turbine-generator speed. The turbine-generator speed is constant and is not dependent upon reactor power level. The UFSAR evaluations and conclusions related to turbine overspeed protection are not affected by the proposed uprated conditions. The existing analysis of this event remains applicable for operation at the uprated conditions.

II.2.15 Steam Generator Tube Rupture - UFSAR 14.3.1

The SGTR accident is discussed in UFSAR Section 14.3.1. The accident analyses demonstrate that the radiological dose consequences are less than the regulatory limits and that SG overfill does not occur for the complete severance of a single SG tube near the top of the tube bundle.

The thermal-hydraulic analysis of record uses the NRC-approved RETRAN analysis methodology (Reference II-17) to predict the break flow in the ruptured SG and the response of the RCS and secondary system. RETRAN also calculates the fraction of the break flow that flashes directly to steam, for use in the dose analysis, and steam releases from the ruptured and intact SGs via the atmospheric steam dumps and main steam safety valves. The analysis assumed a core power of 2597 MWt, or 102% of 2546 MWt (current RP), to generate the steam release rates. Therefore, the analyzed core power to generate the steam releases is bounding for the MUR power uprate.

No explicit safety analysis is performed to demonstrate that no liquid inventory enters the main steam lines (SG overfill). The basis for having no explicit analysis is industry experience with real SGTR events (Ginna, North Anna, Surry, and Prairie Island) and simulator training exercises that validate emergency operating procedures. Therefore, explicit deterministic analyses to address overfill are not performed. The small increase in core power being proposed will not reduce the effectiveness of the emergency operating procedures in preventing an overfill condition. The SGTR event is bounding for the MUR power uprate.

II.2.16 Rupture of Main Steam Pipe - UFSAR 14.3.2

The rupture of a main steam pipe is analyzed in UFSAR Chapter 14 for DNBR. The calculation uses the NRC-approved RETRAN methodology (Reference II-17) for the transient response and the NRC-approved COBRA methodology with the W-3 correlation (Reference II-13) for the DNBR analysis. Because the steam line break is limiting at 0% core power, the UFSAR analysis is unaffected by the MUR power uprate.

II.2.17 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) - UFSAR 14.3.3

The control rod assembly ejection accident is analyzed for RCS overpressure, PCT, percent fuel melt, and average fuel enthalpy. The analysis of record was consistent with the assumptions, methodology, and calculational techniques of the NRC-approved rod ejection topical report (Reference II-19).

Rod ejection analyses are performed at HZP and HFP. A point kinetics RETRAN analysis is performed at nominal HFP conditions and a hot spot RETRAN analysis is performed at deterministic HFP conditions (102% of RP). The hot spot analysis model used a nominal core power of 2546 MWt with an FQ of 2.397 that included the 2% power calorimetric uncertainty to determine the initial hot spot model power level. Thus, the hot spot analysis accounts for 2% power uncertainty above the current RP of 2546 MWt, and the analyzed core power is bounding for the MUR power uprate. Analyses at HZP are not affected by the MUR power uprate.

UFSAR Section 14.3.3.2.1.3 describes the RCS overpressure analysis. Section 2.2.5 of VEP-NFE-2-A (Reference II-19) refers to a very conservative, generic overpressure analysis that was performed by Westinghouse in WCAP-7588, Revision 1-A (Reference II-20). Sections 2.6 and 4.4 of WCAP-7588, Revision 1-A, describe the methodology and conservative analysis. A review of the reactivity assumptions in the generic analysis and the Surry rod ejection analyses indicates that the generic analysis remains bounding for the Surry MUR uprate condition. Therefore, the rod ejection accident is bounding for the MUR power uprate.

II.2.18 Fuel Handling Accidents - UFSAR 14.4.1

The current fuel handling accident radiological analysis is discussed in UFSAR Section 14.4.1 and is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The core inventory source term used in the current fuel handling accident analysis is a function of core power, enrichment, burnup, gap fractions, the number of failed fuel rods, and the assumed radial peaking factor. The existing fuel handling accident dose evaluation was performed using a core inventory that assumes 2605 MWt, which is 102.3% of 2546 MWt, and a single failed fuel assembly (204 rods). No changes to the assumed number of failed fuel rods, assumed radial peaking factor, or gap fractions are required to support the MUR power uprate. The continued applicability of the gap fractions is verified each cycle, as part of the cycle reload safety evaluation process, to confirm the assumed gap fractions remain bounding for each specific core design. The release pathways, χ/Qs, and dose conversion factors are unchanged from the AST license amendment requests and associated SERs (References II-26 and II-27). Therefore, the current fuel handling accident dose evaluation remains bounding for the MUR power uprate.

II.2.19 Volume Control Tank Rupture - UFSAR 14.4.2.1

The volume control tank (VCT) rupture analysis was submitted as part of the FSAR and subsequently updated under the provisions of 10 CFR 50.59. The VCT rupture dose analysis is described in UFSAR Section 14.4.2.1 and is based on the guidance from NRC Branch Technical Position ETSB 11.5, Revision 0. The analysis assumes a source term corresponding to 1% failed fuel fission product inventory in the RCS and results in an EAB whole body dose of less than 0.5 Rem. Replacing the current 1% failed fuel inventory with the revised MUR 1% failed fuel inventory (See Section III.2) would result in a decrease in dose from a VCT rupture. This decrease in the dose is largely due to the smaller Kr-88 and Xe-133 content in the MUR 1% failed fuel inventory relative to the current 1% failed fuel inventory as can be seen in Table II-4. Based on the evaluation of source term change, the current VCT rupture radiological consequence analysis is bounding for the MUR power uprate.

Table II-4
1% Failed Fuel RCS Noble Gas Concentration Comparison
Current versus Revised MUR

Nuclide	Current 1% FF Tech Spec RCS Concentration (μCi/gm)[1]	Revised MUR 1% FF Tech Spec RCS Concentration (μCi/gm)	Dose Difference (Revised - Current)[2] (mRem WB)
Kr-85m	1.52E+00	1.12E+00	-1.6
Kr-85	3.23E+00	4.52E+00	0.1
Kr-87	1.04E+00	7.12E-01	-2.5
Kr-88	3.75E+00	2.02E+00	-66
Xe-133m	2.49E+00	3.08E+00	0.5
Xe-133	2.51E+02	1.96E+02	-61
Xe-135m	1.73E-01	7.04E-01	1.1
Xe-135	6.93E+00	7.40E+00	2.8
Xe-138	4.67E-01	5.08E-01	0.2

1. Values from UFSAR Table 9.1-4 in μCi/cc converted to μCi/gm with a density of 0.75 gm/cc.
2. Using RG 1.109 dose conversion factors.

II.2.20 Waste Gas Decay Tank Rupture - UFSAR 14.4.2.2

The waste gas decay tank (WGDT) rupture analysis was submitted as part of the FSAR and approved in the associated SER (Reference II-29). The WGDT analysis was subsequently updated under the provisions of 10 CFR 50.59. The WGDT rupture was analyzed consistent with the activity limit defined in Technical Specification 3.11.B. The result of the analysis was an EAB whole body dose of less than 0.5 Rem, which is reported in UFSAR Section 14.4.2.2 and is consistent with the acceptance criteria of NRC Branch Technical Position ETSB 11.5, Revision 0. The Technical Specification 3.11.B limit on tank activity ensures that the impact of the MUR remains bounded by the current analysis.

II.2.21 Radioactive Liquid Release - UFSAR 14.4.3

This UFSAR section merely describes the various plant design features used to prevent a liquid release impacting offsite dose levels. There is no analysis associated with this section.

II.2.22 Major Reactor Coolant Pipe Rupture (Large Break Loss-of-Coolant Accident) - UFSAR 14.5.1

UFSAR Section 14.5.1 describes the large break LOCA analysis for the Westinghouse SIF product. The analysis applies the NRC approved Westinghouse ASTRUM Best-Estimate LOCA (BELOCA) analysis methodology described in Reference II-4 for calculation of PCT and oxidation (local and whole-core). The NRC approved the use of the ASTRUM methodology for Surry BELOCA analysis in Reference II-5. Since NRC approval, the BELOCA analysis has been augmented by evaluations under 10 CFR 50.46. The analysis of record uses a core power of 2597 MWt, which is 102% of 2546 MWt, with no additional core power uncertainty applied. Therefore, the analyzed core power is bounding for the MUR power uprate.

UFSAR Section 14.5.1.6 concludes that the LOCA long-term core cooling requirement of 10 CFR 50.46(b)(5) is met. Implicit in that conclusion is the acceptability of the ECCS long-term water supply to the core and the procedures to mitigate the build-up of boric acid in the core. The analysis of record for post-LOCA containment sump boron concentration (subcriticality), containment sump pH, and hot leg switchover time were reviewed for the MUR power uprate. The most recent NRC approval (Reference II-9) of these analyses was in a license amendment that increased the RWST and accumulators boron concentration limits. These analyses have been supplemented by additional evaluations performed under the provisions of 10 CFR 50.59. The following evaluations confirm that the analyses of record remain bounding for the proposed MUR power uprate, and that long-term cooling is assured.

- The containment sump pH calculation does not explicitly include a core power level. The methodology normalizes the contributing inventories to a sump temperature of 70°F. The proposed core power increase does not affect the analysis that determines the post-LOCA sump pH.
- The minimum containment sump boron concentration calculation that ensures post-LOCA subcriticality does not explicitly include a core power level. Each core reload confirms that the post-LOCA sump boron concentration provides adequate subcriticality during the vessel reflood stage, the switchover to cold

leg recirculation, and during long-term core cooling. The core reload confirmation of post-LOCA sump boron concentration limits accounts for the core power level.

- The hot leg switchover time calculation uses a core power level of 2597 MWt, or 102% of 2546 MWt, to determine the post-LOCA core steaming rate. This analysis remains bounding for the proposed MUR power uprate.

II.2.23 Loss of Reactor Coolant From Small Ruptured Pipes or Cracks in Large Pipes, Which Actuates the Emergency Core Cooling System (Small Break Loss-of-Coolant Accident) - UFSAR 14.5.2

UFSAR Section 14.5.2 describes the SBLOCA analysis for the Westinghouse SIF product. The analysis applies the NRC approved methodologies in References II-6 and II-7 for calculation of PCT and oxidation (local and whole-core). The NRC approved the use of the methodologies for Surry SBLOCA analysis in Reference II-8. Since NRC approval, the SBLOCA analysis has been augmented by reanalyses and evaluations under 10 CFR 50.46. The analysis of record uses a core power of 2597 MWt, which is 102% of 2546 MWt. Therefore, the analyzed core power is bounding for the MUR power uprate.

II.2.24 Radiological Consequences

II.2.24.1 LOCA Dose Evaluations

As discussed in UFSAR Section 14.5.5, the LOCA event analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The existing SPS Technical Specification 5.3.1 restricts fuel enrichment to 4.3 w/o U-235, which is unchanged by the MUR power uprate. Fuel assembly exposure is restricted to a lead rod burnup of 62,000 MWD/MTU. The MUR power uprate results in limited changes to core power, burnup history, and enrichment. Thus, the source term core inventory incorporated into the existing LOCA dose analysis remains bounding.

The current LOCA dose analysis is based on a core inventory that assumes 2605 MWt, which is 102.3% of 2546 MWt. The LOCA radiological consequences result from the release of the core inventory to the RCS and then to the environment. The release pathways, χ/Qs, and dose conversion factors are unchanged from the AST and Generic Safety Issue 191 license amendment requests and associated SERs (References II-2, II-3, and II-26 through II-28). Therefore, the existing LOCA radiological analysis remains bounding for the MUR power uprate.

II.2.24.2 Locked RCP Rotor Dose Evaluation

As discussed in UFSAR Section 14.2.9.2.4, the LRA analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The LRA analysis was submitted as part of the stretch power uprate license amendment request and approved in the associated SER (References II-18 and II-8). The LRA analysis was subsequently updated to AST methods under the provisions of 10 CFR 50.59 after the approval of the AST in Reference II-27.

The core inventory source term used in the current LRA analysis is a function of core power, enrichment, burnup, gap fractions for non-LOCA events from Regulatory Guide 1.183, an assumed percent of failed fuel, and an assumed radial peaking factor. The existing LRA dose evaluation was performed using the core inventory that assumes 2605 MWt, which is 102.3% of 2546 MWt. No changes to the assumed percent of failed fuel or assumed radial peaking factor are required to support the MUR power uprate. The steam release modeled in the current LRA analysis is consistent with a core thermal power of 2596.9 MWt (102% of 2546 MWt). The release pathways, χ/Qs, and dose conversion factors are unchanged as a result of the MUR. Therefore, the current LRA dose evaluation remains bounding for the MUR power uprate.

II.2.24.3 Fuel Handling Accident Dose Evaluation

The fuel handling accident dose evaluation is discussed in Section II.2.18.

II.2.24.4 Main Steam Line Break Dose Evaluation

The current main steam line break (MSLB) radiological analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The MSLB analysis was submitted as part of the stretch power uprate license amendment request and approved in the associated SER (References II-18 and II-8). The MSLB analysis was subsequently updated to AST methods under the provisions of 10 CFR 50.59 after the approval of the AST in Reference II-27. The analysis involves the release of contaminated primary coolant through the SG to the environment. Changes to the MSLB dose analysis related to the revised RCS source term that is required to support the MUR are discussed in Section III.2.B.

II.2.24.5 Steam Generator Tube Rupture Dose Evaluation

The current steam generator tube rupture (SGTR) radiological analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The SGTR analysis was submitted as part of the stretch power uprate license amendment request and approved in the associated SER (References II-18 and II-8). The SGTR analysis was subsequently updated to AST methods under the provisions of 10 CFR 50.59 after the approval of the AST in Reference II-27. The analysis involves the release of contaminated primary coolant through the SG to the environment. Changes to the SGTR dose analysis related to the revised RCS source term that is required to support the MUR are discussed in Section III.2.B.

II.2.24.6 Volume Control Tank Rupture Dose Evaluation

The volume control tank rupture dose evaluation is discussed in Section II.2.19.

II.2.24.7 Waste Gas Decay Tank Rupture Dose Evaluation

The WGDT rupture dose evaluation is discussed in Section II.2.20.

II.2.25 Natural Circulation - UFSAR 14.2.12

Natural circulation is analyzed in two events: LOOP and Appendix R safe shutdown.

In the event of a complete LOOP and a turbine trip, there will be a loss of power to the plant auxiliaries (the RCPs, condensate pumps, etc.). The events following a loss of AC power with turbine and reactor trip are described in Section 14.2.12 of the UFSAR. The main difference between this event and the loss of normal feedwater event in Section II.2.12 is that the loss of power trips the RCPs, which reduces the primary system heat load but requires natural circulation cooling. The loss of all AC power case is bounded by the loss of normal feedwater analysis in UFSAR Section 14.2.11, which was analyzed at 102% of 2546 MWt core power. The analysis is bounding for the MUR power uprate.

The SPS Appendix R Report, Section 3.5.1 states that one RCS loop is required to ensure that natural circulation can be established and maintained. A review of safety analysis calculations referenced in the SPS Appendix R Report, confirmed that the safe shutdown systems provide adequate natural circulation cooling after the MUR power uprate.

II.2.26 LOCA Mass and Energy Release

II.2.26.1 Long-term LOCA Mass and Energy Release Analysis - UFSAR 5.4.1

The long-term LOCA mass and energy release analysis methodologies used in the UFSAR Chapter 5 and 6 containment analyses were submitted to the NRC in Reference II-2. The NRC approved these LOCA containment analyses in Reference II-3. Westinghouse mass and energy release analyses for the blowdown and reflood phases used NRC approved methods and assumed a core power of 102% of 2546 MWt. The GOTHIC post-reflood mass and energy releases were generated with NRC approved methods, assuming a core power of 102% of 2546 MWt.

Subsequent to the NRC approval in Reference II-3, SPS performed a reanalysis of the long-term LOCA mass and energy releases. The revised containment analysis used the same NRC approved methods described in the UFSAR and approved in Reference II-3, assumed a core power of 102% of 2546 MWt, and was implemented under the provisions of 10 CFR 50.59. The LOCA mass and energy releases remain bounding for the MUR power uprate conditions. Since the MUR power uprate has no effect on the containment heat sinks, free volume, or heat removal systems and a conservative core power level was used for the generation of mass and energy releases, the UFSAR LOCA containment response analyses remain bounding for the MUR power uprate.

II.2.26.2 Short-term LOCA Mass and Energy Release Analysis - UFSAR 15.6.3

UFSAR Section 15.6.3 includes a brief description of analyses of containment subcompartment response post-LOCA. The short-term LOCA mass and energy releases are affected by changes in RCS temperatures due to the fluid density effect on the initial pressure pulse created when the pipe ruptures. The power uprate design RCS temperatures were reviewed and confirmed to be bounded by the existing evaluations for subcompartment structures, including the pressurizer cubicle, SG cubicle and reactor vessel cavity (Reference II-8, SER Section 3.3.5). Therefore, the containment subcompartment analyses remain bounding for the MUR power uprate.

II.2.27 Main Steam Line Break Mass and Energy Release - UFSAR 5.4.3

The MSLB mass and energy releases used in the UFSAR Chapter 5 containment analyses were described in Attachment 1, Section 3.2.2 of license amendments supporting the resolution of NRC Generic Safety Issue 191 (Reference II-2). The MSLB mass and energy release analyses used NRC approved methods. The full-power analyses assumed a core power of 2957 MWt, which is 116.1% of 2546 MWt. The data was generated for North Anna Power Station, which has a higher core thermal power than SPS. In Reference II-3, the NRC approved the MSLB containment analyses and the use of North Anna MSLB mass and energy releases submitted in Reference II-2. The MSLB mass and energy releases remain bounding for the MUR power uprate conditions. The UFSAR MSLB containment response analyses remain bounding, because the power uprate has no effect on the containment heat sinks, free volume or heat removal systems, and a conservative core power level was used to generate at-power mass and energy releases.

II.2.28 ATWS/AMSAC - UFSAR 7.2.3.2.5 and 7.2.3.2.7

In compliance with 10 CFR 50.62, *Requirements for Reduction of Risk from Anticipated Transient Without SCRAM (ATWS) Events for Light-Water-Cooled Nuclear Power Plant,* ATWS mitigation circuitry has been incorporated into the Surry Units 1 and 2 plant design. The purpose of the Anticipated Transient Without Scram (ATWS) Mitigation System is to automatically initiate a turbine trip and auxiliary feedwater (AFW) start under conditions indicative of an ATWS and a loss of feedwater. The Surry AMSAC system was described in a submittal to the NRC in Reference II-11. Section F, *Operating Bypasses,* in Reference II-11 confirmed the applicability of the generic analyses in NS-TMA-2182 (Reference II-10) to Surry. The NRC approved the Surry AMSAC design in Reference II-12.

The AMSAC system and the analyses in NS-TMA-2182 (Reference II-10) were reviewed with respect to the proposed power uprate. Surry is a 3-loop PWR with Model 51F steam generators that is very similar to the 3-loop plant model that was analyzed in NS-TMA-2182 (Reference II-10). The key differences are core power and total primary system relief valve capacity (pressurizer safety valves and power operated relief valves). The generic plant analyses in Reference II-10 assumed a core power of 2785 MWt, which is 109.4% of the current Surry core rated power of 2546 MWt. Therefore, the generic analysis power level is bounding for Surry. The total RCS relief capacity in the generic analysis is slightly greater than Surry's relief capacity. Generic analyses in Reference II-10 at 2785 MWt core power with one pressurizer PORV failure represents a bounding configuration for Surry's core power and total RCS pressure relief capacity and showed at least 250 psi margin to the acceptance criterion of 3200 psig. It was concluded that the generic analyses in Reference II-10 are bounding for Surry at 2597 MWt (102% of 2546 MWt) core power. Based on a review of the generic AMSAC System design basis, it is concluded that the existing AMSAC design is adequate for MUR uprate conditions.

The Surry AMSAC design specifies a nominal permissive (C-20) setpoint based on the generic setpoint of 40% turbine load minus an allowance for channel inaccuracies in the turbine impulse pressure channels. Some instrument rescaling and calibration will be required for the main turbine first stage pressure input to the AMSAC due to a higher full-power steam flow resulting from the MUR power uprate. There are no other impacts on the AMSAC as a result of changes associated with the proposed power uprate.

II.2.29 Station Blackout - UFSAR 8.4.6

SBO is discussed in Sections V.1.B.i and V.1.B.ii.

II.2.30 Analyses to Determine EQ Parameters - UFSAR 7.5.3.5

Critical EQ parameters include temperature, pressure, radiation, relative humidity, pH and submergence. Current analyses for long-term LOCA and steamline break mass and energy releases remain bounding at the power uprate conditions. The temperature, pressure, relative humidity, pH and submergence conditions are therefore bounding for the proposed uprate. Radiological doses used in the EQ evaluations do not bound the increase in doses due to the power uprate. Therefore, Dominion concludes that, with the exception of radiological doses, the current EQ parameters remain bounding for the MUR power uprate. The evaluation for radiological effects is discussed in Section III.1.

II.2.31 Safe Shutdown Fire Analysis (Appendix R Report) - UFSAR 9.10

UFSAR Section 9.10 describes the fire protection system and design bases for compliance with 10 CFR 50, Appendix R. The SPS Appendix R Report describes the system functions that ensure safe shutdown is achieved after a fire. SPS Appendix R Report, Section 3.8 identifies the calculations that provide the technical basis for the Appendix R fire protection program. Reviews concluded that the calculations cited in the Appendix R Report remain bounding for the MUR power uprate. Additional calculations that are not cited in the Appendix R Report, but have provided a basis for the program were reviewed and remain bounding for the MUR power uprate.

The safe shutdown analyses that support the Appendix R program were reviewed. The analyses support a core power of 2597 MWt, or 102% of the current RP of 2546 MWt. The power uprate does not change the design, function or impose any new requirements on the systems or components that support the Appendix R safe shutdown requirements (e.g., RHR, chemical and volume control). Operator actions in response to an Appendix R fire are not adversely impacted. The MUR power uprate does not affect the worst case fire location or the post-fire local operations and capability to complete repairs. The worst case fire scenario timeline indicates that the plant can achieve cold shutdown within the 72-hour requirement. The 72-hour cooldown requirement in 10 CFR 50, Appendix R, Sections III.G.1.b and III.L is met. Therefore, the Appendix R safe shutdown analyses remain bounding for the MUR power uprate.

II.2.32 Spent Fuel Pool Cooling - UFSAR 9.5

The SPS UFSAR outlines the cooling requirements for the SFP. Two scenarios are described. Each scenario assumes that fuel movement begins no earlier than 100 hours after the reactor is subcritical. First – normal back-to-back scenario, where one unit's full core offload follows immediately after the other unit's refueling. Second – abnormal back-to-back scenario, where an emergency full offload occurs after two back-to-back refueling outages. The design basis SFP temperature limit for the first scenario is 140°F; the second scenario limit is 170°F. The SFP heat loads in the analyses of record were calculated to include 2% instrument uncertainty. The maximum MUR heat load is bounded by the calculated maximum value. The SFP cooling system is capable of maintaining the SFP pool temperature below the design basis limits with the above heat loads. Therefore, there is no change to the loss of cooling analysis.

The spent fuel pool heat load assumptions included in the SPS UFSAR will remain bounding for the MUR power uprate. Refer to Section VI.1.D for further discussion on the spent fuel pool storage and cooling.

II.2.33 Internal Flooding - UFSAR 2.3.1.2

The design bases for flooding inside and outside the containment building were evaluated. The power uprate results in increased piping system flowrates (e.g., condensate, main feedwater and main steam). These changes were evaluated to determine any impact on the flooding analysis. Based on flooding analysis calculation reviews, it was determined that the current flood levels are not affected by the MUR power uprate.

II.2.34 Transient Analysis of a High-Energy Line Break in the Main Steam Valve House - UFSAR 14B.6

UFSAR Section 14B.6 describes a special analysis case for a loss of all feedwater from a HELB in the MSVH. The HELB event is assumed to disable all AFW pumps in the MSVH, and the recovery of a secondary heat sink is provided by AFW from the opposite unit. The analysis described in UFSAR Section 14B.6 supports the AFW system cross-connect requirements in Surry Technical Specification 3.6. The plant transient response was performed using the NRC-approved RETRAN analysis methodology (Reference II-17) with an initial reactor power of 2597 MWt, or 102% of 2546 MWt. The analysis specifies two operator actions that must be performed within 10 minutes of the accident initiation. Adequate AFW flow must be provided from the opposite unit and the reactor coolant pumps must be tripped. Because these requirements are based on an analysis at 102% of 2546 MWt core power, the operator actions are not affected by the MUR power uprate. In conclusion, the analysis of the HELB in the MSVH is bounding for the MUR power uprate.

II.2.35 Rupture of a Main Feedwater Pipe (not described in UFSAR)

A major feedwater line break (MFLB) is defined as a break in a large feedwater pipe which interrupts the addition of main feedwater to the steam generators and results in the discharge of secondary inventory from the affected steam generator to the containment. A MFLB is a loss of heat sink transient but can have some of the characteristics of a steam line break, i.e., an initial cooldown of the RCS resulting from the secondary depressurization.

The MFLB event is not a UFSAR Chapter 14 event for Surry, which was licensed prior to the issuance of Regulatory Guide 1.70, Revision 1 (Reference II-21). The MFLB analysis was performed to develop conditions at the inlet to the pressurizer safety and relief valves in response to NRC questions related to NUREG-0737. The analysis was described as a special event in Section 3.8.1 of Attachment 3 of the license amendment for the core uprate to 2546 MWt (Reference II-18), which was approved by the NRC in Reference II-16. For the MUR power uprate, the MFLB analysis was reviewed and confirmed to be analyzed at a core power of 102% of 2546 MWt. Therefore, the analysis remains bounding for the MUR power uprate.

II.3 Design Transients

II.3.1 Nuclear Steam Supply System Design Transients

NSSS design transients were specified in the original design analyses of NSSS components cyclic behavior. The selected transients are conservative representations of transients that when used as a basis for component fatigue analysis, provide confidence that the component is appropriate for its application over the 60-year plant license period. The RCS and its auxiliary system components are designed to withstand the cyclic load effects from RCS temperature and pressure changes. The existing design transients were evaluated for their continued applicability at MUR power uprate conditions.

The key plant design parameters for the NSSS design transients are RCS hot and cold leg temperatures (T_{hot}, T_{cold}), secondary side steam temperature and pressure (T_{steam}, P_{steam}), and the secondary side feedwater temperature. The original design transients did not include the feedwater temperature and flow responses. These were developed for any design transient that required modification due to the MUR power uprate. Note that the previous design transient analysis was generic, and was revised to represent a SPS specific analysis. Some conservatism was removed from the existing design transients so they would better represent uprated plant conditions. The new design transients have been developed and the results of the component fatigue analyses were acceptable. The revised design transients were considered in the various NSSS component evaluations to ensure the component fatigue analyses were satisfactory. The component fatigue evaluation results are discussed in Section IV.

The primary to secondary differential pressure limit was not exceeded for any normal or upset design transient. The frequencies of occurrence for the 60-year plant licensed period are unchanged for the power uprate. No new design transients are created as a result of the MUR power uprate.

II.3.2 Auxiliary Equipment Design Transients

The SPS auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the RCS and certain NSSS auxiliary systems piping, heat exchangers, pumps and tanks as applicable. The transients are sufficiently conservative, such that when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the plant operating license period.

The only auxiliary equipment design transients potentially impacted by the power uprate are those transients associated with full load RCS design temperatures (T_{hot} and T_{cold}). These temperature transients are defined by the differences between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. Since the operating coolant temperatures in the auxiliary systems are not impacted by the power uprate, the temperature difference between auxiliary systems and the RCS loops is only affected by changes in the RCS operating temperatures. The design temperature transients assume a full load T_{hot} and T_{cold} of 630°F and 560°F, respectively. These full load temperatures were selected for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters. The approved NSSS design temperature range for T_{hot} and T_{cold} used to develop the current design temperature transients is smaller than the reference design temperature values. The smaller full load temperatures from the MUR power uprate result in less severe design temperature transients. Therefore, the existing auxiliary equipment design transients are conservative and bounding for the MUR power uprate.

The design transients evaluated for the MUR are thermal transients associated with the difference between the design full power values of T_{hot} and T_{cold}. Magnitude of the transients is defined by the difference between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. This analysis did not include the potential impact of changes in nitrogen-16 activity and its relationship with letdown line decay time requirements. This aspect of the MUR is discussed below.

The existing design basis requirement is that coolant flow leaving the RCS loop through the letdown line has a transit time of at least 60 seconds to reach the containment penetration (assuming maximum letdown flow). This delay time is required to allow for decay of nitrogen-16. The delay depends on two key design features: (1) the letdown flowrate and (2) the total volume of piping through which flow passes between the RCS loop and the containment penetration. The letdown line decay requirement does not have a dependence upon reactor power. Since the MUR does not affect the letdown flowrate or letdown piping volume, the specified design requirement remains satisfied for operation at the proposed MUR conditions.

II.3.3 Plant Operability

The pressure control component sizing and plant operability for normal condition transients were evaluated for SPS.

RCS pressure control component sizing includes the pressurizer heater, spray, and PORV capacities. These components must continue to successfully perform their intended functions. Plant operability for Condition I (normal condition) transients includes the plant response to 5-percent/minute loading and unloading, 10-percent step-load increase or decrease, and large-load rejection. These transients must not result in a reactor trip, engineered safeguards features actuation or challenge the pressurizer or main steam safety valves. This evaluation was conducted to confirm the continued plant acceptability to meet these requirements at power uprate conditions.

Pressure control component sizing and plant operability for normal condition transients were reviewed independently. The reviews concluded that the power uprating does not result in unacceptable plant operations. The existing pressure control components (heater, spray, and PORV) meet the sizing criteria at the uprated conditions. The component capacities are adequate to mitigate the sizing basis transients without exceeding the limits. Adequate margin exists to relevant reactor trip and engineered safeguards features actuation system setpoints during the normal condition transients at uprated power conditions. The control systems remain stable and support the power uprate for normal condition transients. The existing setpoints for the reactor control, pressurizer pressure control, pressurizer level control, SG level control, and steam dump control remain valid.

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III. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling and flooding).

2. For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:

   A. Identify the transients/accidents that is the subject of the analysis.

   B. Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate.

   C. Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59.

   D. Provide a reference to the NRC's approval of the plant's reload methodology.

3. For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should include:

   A. Identify the transient or accident that is the subject of the analysis.

   B. Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate.

   C. Confirm that the limiting event determination is still valid for the transient or accident being analyzed.

   D. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies.

E. Provide references to staff approvals of the methodologies in Item D above.

F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology.

G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate.

H. Describe and justify the chosen single-failure assumption.

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate.

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head (NPSH), valve isolation capabilities) required to support the analysis.

K. Discuss the results and acceptance criteria for the analysis, including any changes from previous analysis.

RESPONSE TO III - ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

III.1 Analysis to Determine EQ Parameters Radiological Effects - UFSAR 7.5.3.5

III.1.A Normal Operation

Normal non-radiological plant operating conditions assumed within all environmental zones (i.e., temperature, pressure, humidity) remain unchanged for power uprate operation. A separate evaluation was performed to assess potential increase in normal operation radiation dose used in the EQ program. In general, power uprate operation would be expected to increase the core inventory of radioisotopes by the percentage increase in core power and potentially to increase the normal operation radiation source term. However, this potential increase in radiation source term will not affect the currently estimated normal operation doses used for EQ, because of several conservative factors incorporated into the current estimates. The most significant of these considerations are: (a) use of a dose for a given radiation zone designation that represents the maximum end of the normal operation range and (b) the limitation imposed by plant operations as a result of Technical Specifications limits on RCS coolant activity (i.e., allowable limits of operation are approximately one-third of the value associated with the assumed 1% fuel defects used in the normal operation source term). The conditions used in the EQ program for normal operation therefore remain bounding for the MUR power uprate, with the exception of the dose levels for the reactor vessel excore neutron detectors. The excore detectors radiation dose increases such that the EQ in-service life may be decreased. These excore detectors are scheduled to be replaced on Unit 1 in Fall 2010 and Unit 2 in Spring 2011. Prior to operating above 2546 MWt (98.4% RP), Dominion will determine the EQ service life of the excore detectors. A calculation is being developed to evaluate the dose impact on these detectors. Based on results of the comparable North Anna calculation, there is no anticipated impact on radiation dose margin or qualified life.

III.1.B Accident Conditions

There is no change in assumed accident temperature, pressure, or humidity due to power uprate operation. The post accident (i.e., LOCA) radiation effects have been updated to reflect the power uprate conditions. The evaluation details are provided below.

The current post accident dose estimates utilized for EQ are based on LOCA and radiation source terms corresponding to a core power level of 2546 MWt, assuming a 12-month fuel cycle and the ACTIVITY2 computer code. These were the design basis calculations from original plant licensing. For the MUR power uprate, the applicable assumptions of the post-accident radiation source terms are a core power of 2597 MWt and 18-month fuel cycle. The computer code used to develop the core inventory applicable for the MUR uprate is ORIGEN2.

In Section 2.3.1 of the Millstone 3 Stretch Power Uprate license amendment request (Reference III-1) scaling factors were developed that accounted for an increase in core power (3636 to 3723 MWt), an 18-month versus 12-month fuel cycle, and current use of the ORIGENS computer code versus the ACTIVITY2 code used in the original design basis analysis. For the SPS MUR power uprate, an evaluation was performed to confirm applicability of the scaling factors developed for Millstone 3 to SPS and that they are conservative. The resulting factors were used to modify the existing post-accident total integrated radiation dose for all environmental zones identified in the SPS EQ program. These augmented values represent the MUR power uprate radiation environment considered for EQ. Table III-1 provides a summary of the current and revised radiation parameters, for each separate class of equipment that is monitored within the EQ program.

Table III-1
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
15.1	12	CONAX BUFFALO CORP.	7100-10000 7115-10000 7737-10000 series	RC-18B	5.25E+04	7.40E+06	1.69E+07	2.44E+07	8.88E+06	2.58E+07	1.00E+08	74%
15.5	9	AMPHENOL	Types IA, IB, IC, III, IVA, IVB	RC-18B	5.25E+04	7.40E+06	0.00E+00	7.45E+06	8.88E+06	8.93E+06	1.00E+07	11%
15.6	2	WESTINGHOUSE ELECT CORPORATION	WX35040	RC-18B	5.25E+04	7.40E+06	0.00E+00	7.45E+06	8.88E+06	8.93E+06	1.60E+08	94%
16.1	4	RAYCHEM CORP	HVT	AB-13D	4.05E+07	9.30E+05	0.00E+00	4.14E+07	1.12E+06	4.16E+07	1.00E+08	58%
16.2	11	RAYCHEM CORP	WCSF/NJRT	RC-3A	1.95E+07	2.40E+07	8.45E+07	1.28E+08	2.88E+07	1.33E+08	2.15E+08	38%
16.5	6	RAYCHEM CORP	NMCK8	AB-2C	4.20E+06	8.00E+06	0.00E+00	1.22E+07	9.60E+06	1.38E+07	5.00E+07	72%
16.6	1	SCOTCH 3M	VARIOUS	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	1.00E+08	52%
17.1	9	BUCHANAN, GENERAL ELECTRIC, MARATHON	VARIOUS	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
17.2	9	BUCHANAN, GENERAL ELECTRIC, MARATHON	VARIOUS	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
3.1	25	LIMITORQUE	SB, SMB, SMC, SBD	AB-2C	4.20E+06	8.00E+06	0.00E+00	1.22E+07	9.60E+06	1.38E+07	2.00E+07	31%

Table III-1 (Continued)
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
3.2	9	LIMITORQUE	SMB	RC-241A RC-262A RC-291A	4.50E+07	1.80E+07	0.00E+00	6.30E+07	2.16E+07	6.66E+07	2.04E+08	67%
34.1	8	CONAX BUFFALO CORP.	N-11000 SERIES	SUM-1	1.95E+08	6.80E+02	0.00E+00	1.95E+08	8.16E+02	1.95E+08	2.25E+08	13%
34.2	8	CONAX BUFFALO CORP.	PL SERIES/4P	SFGD-1	1.32E+03	8.00E+06	0.00E+00	8.00E+06	9.6E+06	9.60E+06	2.25E+08	96%
34.2	8	CONAX BUFFALO CORP.	PL SERIES/ VITON	SFGD-1	1.32E+03	7.41E+06	0.00E+00	7.41E+06	8.89E+06	8.89E+06	1.00E+07	12%
34.3	5	ROSEMOUNT, INC.	353C	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	1.00E+08	52%
34.4	6	EGS, A DIVISION OF SAIC	EGS/PATEL QDC	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
34.5	6	EGS, A DIVISION OF SAIC	K-III	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
35.1	31	AUTOMATIC SWITCH CO/ASCO	206-380 206-381 NP-8320 NP-8316 NP-8321 NP-8344	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.05E+08	76%
35.3	10	VALCOR ENGINEERING CORP	V526 SERIES	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
35.4	10	TARGET ROCK	79AB-008	RC-3A	1.95E+07	2.40E+07	6.04E+06	4.95E+07	2.88E+07	5.43E+07	1.85E+08	71%

Table III-1 (Continued)
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
35.6	8	TARGET ROCK	86V-001	RC-3A	1.95E+07	2.40E+07	1.78E+06	4.53E+07	2.88E+07	5.01E+07	1.35E+08	63%
4.1	9	WESTINGHOUSE ELECT CORPORATION	68F13318	AB-2C	4.20E+06	8.00E+06	0.00E+00	1.22E+07	9.60E+06	1.38E+07	5.12E+07	73%
4.2	12	GENERAL ELECTRIC CO.	5K6287XH41A	SFGD-1	1.32E+03	8.00E+06	0.00E+00	8.00E+06	9.60E+06	9.60E+06	4.60E+07	79%
4.3	10	WESTINGHOUSE ELECT CORPORATION	VARIOUS	SFGD-1	1.32E+03	8.00E+06	0.00E+00	8.00E+06	9.60E+06	9.60E+06	1.40E+07	31%
4.4	11	GENERAL ELECTRIC CO.	5K6319XJID	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
4.5	4	CRANE ELECTRIC COMPANY	GA-1K751H-1CA20	CSPH-11	1.32E+03	5.90E+06	0.00E+00	5.90E+06	7.08E+06	7.08E+06	2.00E+08	96%
4.6	3	RELIANCE ELECTRIC CO.	3996ST	AB-2B	3.75E+06	2.50E+06	0.00E+00	6.25E+06	3.00E+06	6.75E+06	2.04E+08	97%
6.1	5	THE ROCKBESTOS COMPANY	FIREWALL III/PYROTROL III	RC-3A	1.95E+07	2.40E+07	2.23E+07	6.58E+07	2.88E+07	7.06E+07	2.00E+08	65%
6.10	7	OKONITE COMPANY	EPR/NEOPRENE	RC-3A	1.95E+07	2.40E+07	2.23E+06	4.57E+07	2.88E+07	5.05E+07	2.00E+08	75%
6.11	6	RAYCHEM CORP	FMR-XLPE	RC-3A	1.95E+07	2.40E+07	8.45E+06	5.20E+07	2.88E+07	5.68E+07	2.00E+08	72%
6.13	4	BRAND-REX COMPANY	XLPE/CSPE	RC-3A	1.95E+07	2.40E+07	2.23E+07	6.58E+07	2.88E+07	7.06E+07	2.00E+08	65%

Table III-1 (Continued)
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
6.14	2	COLLYER COMPANY	XLPE	AB-13D	4.05E+07	9.30E+05	0.00E+00	4.14E+07	1.12E+06	4.16E+07	2.00E+08	79%
6.15	2	KAISER COMPANY	XLPE	AB-13D	4.05E+07	9.30E+05	0.00E+00	4.14E+07	1.12E+06	4.16E+07	2.00E+08	79%
6.16	8	VARIOUS	VARIOUS	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	1.69E+08	71%
6.17	2	GENERAL ELECTRIC CO.	VULKENE	AB-2C	4.20E+06	8.00E+06	0.00E+00	1.22E+07	9.60E+06	1.38E+07	4.00E+07	66%
6.18	8	OKONITE COMPANY	EPR/CSPE	RC-3A	1.95E+07	2.40E+07	8.45E+05	4.43E+07	2.88E+07	4.91E+07	2.00E+08	75%
6.19	1	THE ROCKBESTOS COMPANY	RADIATION RESISTANT SILICONE RUBBER	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
6.20	0	OKONITE COMPANY	EPR/CSPE	RC-3A	1.95E+07	2.40E+07	8.45E+05	4.43E+07	2.88E+07	4.91E+07	2.00E+08	75%
6.3	4	ANACONDA	EPR/NEOPRENE	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
6.4	4	CONTINENTAL WIRE & CABLE	SILICONE RUBBER	RC-3A	1.95E+07	2.40E+07	2.23E+07	6.58E+07	2.88E+07	7.06E+07	1.00E+08	29%
6.5	4	CONTINENTAL WIRE & CABLE	XLPE/CSPE	RC-3A	1.95E+07	2.40E+07	2.23E+07	6.58E+07	2.88E+07	7.06E+07	1.00E+08	29%
6.7	4	OKONITE COMPANY	XLPE/NEO	RC-3A	1.95E+07	2.40E+07	2.23E+07	6.58E+07	2.88E+07	7.06E+07	2.00E+08	65%
6.9	4	OKONITE COMPANY	EPR/NEOPRENE	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%

Table III-1 (Continued)
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
61.1	7	WESTINGHOUSE ELECT CORPORATION	VPA-6 HREEL	RC-47A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.00E+08	76%
71.1	18	DELPHI CONTROL SYSTEMS, INC	K-III	AB-2B	3.75E+06	2.50E+06	0.00E+00	6.25E+06	N/A	N/A	N/A	(1)
8.1	9	TRANSAMERICA-DE LAVAL/GEMS	XM54853/ 54854	RC-27B	5.25E+04	3.50E+07	0.00E+00	3.51E+07	4.20E+07	4.21E+07	2.00E+08	79%
8.12	9	CONAX BUFFALO CORP.	7C47, 7F45	RC-241A RC-262A RC-291A	4.50E+07	1.80E+07	0.00E+00	6.30E+07	2.16E+07	6.66E+07	2.20E+08	70%
8.16	7	GAMMA METRICS INC	RCS-102									(2)
8.17	24	WEED	VARIOUS	SUM-1	1.95E+08	6.80E+02	0.00E+00	1.95E+08	8.16E+02	1.95E+08	3.03E+08	36%
8.18	4	PYCO CO.	122-3027-6	AB-27B	1.95E+08	6.80E+02	0.00E+00	1.95E+08	8.16E+02	1.95E+08	2.20E+08	11%
8.25	9	TEC	1414	RC-3A	1.95E+07	2.40E+07	1.69E+07	6.04E+07	2.88E+07	6.52E+07	2.22E+08	71%
8.26	8	ROSEMOUNT, INC.	01154 SERIES H	RC-3B	5.25E+04	7.40E+06	0.00E+00	7.45E+06	8.88E+06	8.93E+06	5.55E+07	84%
8.3	33	ROSEMOUNT, INC.	1153D	AB-13D	4.05E+07	9.30E+05	0.00E+00	4.14E+07	1.12E+06	4.16E+07	5.00E+07	17%
8.4	19	ROSEMOUNT, INC.	1153 SERIES B	AB-02C	4.20E+06	8.00E+06	0.00E+00	1.22E+07	9.60E+06	1.38E+07	2.21E+07	38%
8.5	9	WESTINGHOUSE ELECT CORPORATION	2654C65G	RC-241A	4.50E+07	1.80E+07	2.00E+07	8.30E+07	2.16E+07	8.66E+07	1.60E+08	46%

Table III-1 (Continued)
MUR Dose Assessment of EQ Components

SPS QDR	QDR Rev.	Manufacturer	Model Type	EZD Zone	Normal Dose (60 yrs) [rads]	Accident Dose Pre-MUR [rads]	Beta Accident Dose [rads]	TID Dose Pre-MUR [rads]	Accident Dose Post-MUR [rads]	TID Dose Post-MUR [rads]	Vendor's Qualified Dose [rads]	Margin
8.6	10	VICTOREEN	877-1/ 878-1-5	RC-47A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.20E+08	78%
8.9	7	ITT BARTON	752	AB-13A	6.56E+03	3.10E+04	0.00E+00	3.76E+04	3.72E+04	4.38E+04	1.00E+05	56%
9.1	26	NAMCO CONTROLS (DIV OF ACME-CLEVELAND)	EA180 SERIES	RC-3A	1.95E+07	2.40E+07	0.00E+00	4.35E+07	2.88E+07	4.83E+07	2.04E+08	76%
9.5	8	MICRO SWITCH	HDLS-LSYPC4L, HDLS-LSYVC4L	AB-2B	3.75E+06	2.50E+06	0.00E+00	6.25E+06	3.00E+06	6.75E+06	1.00E+07	33%
4.7	6	MARATHON ELECTRIC	VARIOUS	AB-13D	4.05E+07	9.30E+05	0.00E+00	4.14E+07	1.12E+06	4.16E+07	2.00E+08	79%

1. Hydrogen monitors are no longer considered safety related per amendment Nos. 239 and 238 to Renewed Facility Operating License Nos. DPR-32 and DPR-37. As a result, hydrogen monitors will be removed from the SPS EQ program and do not need to be evaluated.
2. Awaiting the final evaluation numbers for Surry's replacement components. Based on analysis for North Anna there will be no issues with margin for replacement equipment scheduled to be installed.

III.2 RCS Coolant Activity Source Term – UFSAR 9.1.2.2

As indicated on Table II-2, the RCS coolant activity source term is not bounding for the proposed MUR operation. The RCS coolant activity source term utilized in current SPS design basis documents is based on a core thermal power of 2546 MWt, a 12 month fuel cycle, and 1% fuel defects. This assumed core power equals the power defined as Rated Power in the Surry 1 and 2 Technical Specifications (TS 1.0.A). The key inputs assumed in generating the current design basis source term are listed in Surry UFSAR Table 9.1-5. This source term is the original design basis source term for Surry. The 2546 MWt core power is the value noted in the original FSAR as the maximum expected rating, which was assumed in original analyses of key plant systems. The stretch power uprating (References III-2 and III-3) that was implemented in 1995 achieved operation at a core power of 2546 MWt, while retaining the original RCS source term. For operation at the proposed MUR conditions, the RCS coolant activity source term was updated to accommodate the core power increase (2605 MWt), current operation with 18-month fuel cycles and 1% fuel defects. The assumption of 18-month operation will increase the inventory of long-lived isotopes in the core and the coolant.

The source term for dose analyses that involve the release of primary coolant activity is determined based upon Technical Specifications. Technical Specification primary coolant activity is calculated by normalizing the design 1% fuel defect reactor coolant activities to concentration levels consistent with the Surry Unit 1 and 2 Technical Specification 3.1.D Dose Equivalent Iodine (DEI) limit of 1 μCi/gm DE I-131, determined to be the most limiting condition of operation. The equilibrium appearance rates of each iodine isotope and spiking concentrations were also adjusted for the revised RCS coolant activity. The source term for noble gases and particulate isotopes is correspondingly normalized to the same limiting Technical Specification limit of 1 μCi/gm DEI equivalent to a fuel defect level of 0.51%. This normalization is consistent with existing design basis methods for RCS coolant determination. Because of this normalization, the assumed reactor power level has only an indirect, second order influence on the RCS source term, primarily through the effect on the relative concentration of specific isotopes.

It should be noted that the Maximum Reactor Coolant Activity Technical Specification (TS 3.1.D) limits are in terms of μCi/cc. Specifying coolant sample concentrations in terms of μCi/cc has the potential to lead to errors when used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) as the answer is dependent on the density of the liquid. Surveillances of the coolant activity limits are performed at room temperature and pressure for which the density of coolant samples is 1 gm/cc. Therefore, the Technical Specification limits discussed in Section III.2 will be in units of μCi/gm instead of μCi/cc.

III.2.A Analysis to Accommodate Revised RCS Coolant Activity Source Term

Dominion evaluated the impact of the revised RCS coolant activity source term on existing radiological analyses for which the source term is a key input. The potentially affected events are: Volume Control Tank rupture, Waste Gas Decay Tank rupture, Steam Generator Tube Rupture and Main Steam Line Break. The discussion in Sections II.2.19 and II.2.20, respectively, confirmed that the VCT rupture and Waste Gas Decay Tank rupture event analyses remain bounding assuming the updated source term. The revised analysis results for SGTR and MSLB are presented in Section III.2.B.

III.2.B Steam Generator Tube Rupture and Main Steam Line Break Dose Evaluation

The current radiological DBA analyses of the steam generator tube rupture (SGTR) and main steam line break (MSLB) are based upon the Alternate Source Term (AST) as defined in NUREG-1465 with acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183. The SGTR and MSLB accidents analyses are very similar. Fuel failure is not predicted in these accidents. Therefore, source terms used are the maximum allowed per Technical Specifications. The current dose consequence evaluations for these accidents were performed using the fission product inventory spectrum generated for equilibrium conditions with one percent failed fuel at a core thermal power of 2546 MWt and normalized to the Technical Specification Dose Equivalent I-131 limit of 1 μCi/gm. The reactor coolant activity has been revised for the MUR to reflect 2605 MWt and the current (18-month cycle) fuel management scheme, to obtain an updated isotopic spectrum distribution for the SGTR and the MSLB analyses. New EAB χ/Q values, previously approved by the NRC (Reference III-4), have been incorporated into the SGTR and MSLB analyses to update them to approved license basis assumptions. In addition, updated steam flows from steam generator relief valves have been incorporated into the MSLB and SGTR dose analyses.

A recent station discovery identified issues related to post-accident performance of the main steam atmospheric relief valves (also known as atmospheric dump valves and main steam PORVs). New steam generator PORV relief flow as a function of steam pressure was developed. Using this new flow data, the capability of the PORVs has been assessed against the required cooldown imposed by the accident analyses. In addition, a higher stuck open PORV flow rate was modeled for the SGTR dose analysis, which is conservative. Revised steam flows at various stages following the steam events have been used together with the revised RCS source term updated to MUR conditions and previously approved EAB χ/Qs to demonstrate the MSLB and SGTR remain within regulatory dose limits.

The SGTR and MSLB analyses are discussed below in Sections III.2.B.1 and III.2.B.2, respectively. Results of these analyses are submitted for NRC review and approval. The effects of only the MUR related changes to these analyses are separately identified in Table III-8 (SGTR) and Table III-10 (MSLB). These tables show that the dose consequence increases resulting from RCS source term updates necessary to accommodate the proposed MUR power increase remain within regulatory limits. Dose increases identified are primarily the result of alkali metal contribution from the revised RCS coolant activity compared to that predicted in the original RCS source term.

III.2.B.1 Steam Generator Tube Rupture - UFSAR 14.3.1

A steam generator tube rupture (SGTR) is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of one of the steam generators (denoted as the affected generator) with an assumed release to the environment through the steam generator Power Operated Relief Valves (PORVs) and the steam generator safety valves. The affected generator discharges steam to the environment for 30 minutes until the generator is isolated. The unaffected generator (two generators modeled as one) discharges steam for a period of 8 hours until the primary system has cooled sufficiently to allow a switch over to the residual heat removal system. Consistent with the current licensing basis, the SGTR analysis was performed assuming both a pre-accident iodine spike and a concurrent accident iodine spike. In addition, both loss-of-offsite power (LOOP) and no loss-of-offsite power conditions were considered.

III.2.B.1.1 SGTR Source Term Definition

Initial radionuclide concentrations in the primary and secondary systems for the SGTR accident must be determined. The thermal hydraulic T/H analysis of the SGTR accident indicates that no fuel rod failures occur. Thus, radioactive material releases were determined by the radionuclide concentrations initially present in primary liquid, secondary liquid, secondary steam, and any releases from fuel rods that failed before the transient. These initial values are the starting point for determining the initial curie input for the LOCADOSE code runs.

Regulatory Guide 1.183 indicates that the released activities should be the maximum allowed by the Technical Specifications. Regulatory Guide 1.183 also dictates that the SGTR accidents consider iodine spiking above the value allowed for normal operations based both on a pre-accident iodine spike and a concurrent accident spike. For Surry, the maximum iodine concentration allowed as the result of an iodine spike is 10 μCi/gm dose equivalent I-131. Regulatory Guide 1.183 defines a concurrent iodine spike as an accident initiated value 335 times the release or appearance rate corresponding to the Technical Specification limit for normal operation (1 μCi/gm DE I-131 RCS TS limit) for a period of 8 hours.

Releases of reactor coolant activity are the maximum allowed by Technical Specifications since there is no fuel failure postulated during a SGTR. The initial secondary side liquid inventory was based on the RCS activity normalized to the 0.1 μCi/gm DE I-131 secondary side coolant activity limit. The initial secondary steam noble gas inventory is found by multiplying the primary system noble gas inventory by the dilution ratio. This dilution ratio is the ratio of the primary to secondary leak rate divided by the steam flow rate (See Table III-6). This assumes that all noble gases are carried through the steam generator with steam flow and pass out the PORVs or safety valves and do not build up in the secondary steam. Table III-2 lists all the primary and secondary radionuclide inventories. The radionuclide inventories in Table III-2 include a significant increase in the primary and secondary quantities of Cs that resulted from the updated RCS source term. The concurrent iodine spike rates are listed in Table III-4, while the pre-accident iodine activity at ten times the Technical Specification RCS iodine activity is shown in Table III-3.

The distribution of the iodine isotopes by physical form was dictated by RG 1.183 to be 97% elemental iodine and 3% organic iodine and no particulate iodine.

The TEDE dose conversion factors used to calculate dose for the SGTR accident are built into the LOCADOSE library, which are consistent with Federal Guidance Report 11 and 12 for the isotopes required by Regulatory Guide 1.183.

Table III-2
Isotopes Activities in RCS and SG

Isotopes	Technical Specification DE I-131 (μCi/gm)	Primary Activity (Ci)	Unaffected SG Liquid Activity (Ci)	Affected SG Liquid Activity (Ci)	Unaffected SG Steam Activity (Ci)	Affected SG Steam Activity (Ci)
Kr-83m	1.56E-01	2.85E+01			2.99E-05	1.49E-05
Kr-85m	5.70E-01	1.04E+02			1.09E-04	5.46E-05
Kr-85	2.31E+00	4.22E+02			4.42E-04	2.21E-04
Kr-87	3.64E-01	6.65E+01			6.97E-05	3.48E-05
Kr-88	1.03E+00	1.88E+02			1.97E-04	9.86E-05
Kr-89	3.02E-02	5.51E+00			5.78E-06	2.89E-06
Xe-131m	1.03E+00	1.88E+02			1.97E-04	9.86E-05
Xe-133m	1.58E+00	2.88E+02			3.03E-04	1.51E-04
Xe-133	1.00E+02	1.83E+04			1.91E-02	9.57E-03
Xe-135m	3.60E-01	6.57E+01			6.89E-05	3.45E-05
Xe-135	3.78E+00	6.90E+02			7.24E-04	3.62E-04

Table III-2 (Continued)
Isotopes Activities in RCS and SG

Isotopes	Technical Specification DE I-131 (µCi/gm)	Primary Activity (Ci)	Unaffected SG Liquid Activity (Ci)	Affected SG Liquid Activity (Ci)	Unaffected SG Steam Activity (Ci)	Affected SG Steam Activity (Ci)
Xe-137	7.65E-02	1.40E+01			1.46E-05	7.32E-06
Xe-138	2.60E-01	4.75E+01			4.98E-05	2.49E-05
I-130	1.94E-02	3.54E+00	1.73E-01	8.67E-02	1.18E-04	5.90E-05
I-131	7.45E-01	1.36E+02	6.66E+00	3.33E+00	4.53E-03	2.26E-03
I-132	3.76E-01	6.86E+01	3.36E+00	1.68E+00	2.29E-03	1.14E-03
I-133	1.23E+00	2.25E+02	1.10E+01	5.50E+00	7.48E-03	3.74E-03
I-134	2.42E-01	4.42E+01	2.16E+00	1.08E+00	1.47E-03	7.35E-04
I-135	7.90E-01	1.44E+02	7.06E+00	3.53E+00	4.80E-03	2.40E-03
Cs-134m	1.94E-02	3.54E+00	1.73E-01	8.67E-02	1.18E-04	5.90E-05
Cs-134	1.35E+00	2.46E+02	1.21E+01	6.03E+00	8.21E-03	4.10E-03
Cs-136	2.69E-01	4.91E+01	2.40E+00	1.20E+00	1.63E-03	8.17E-04
Cs-137	8.67E-01	1.58E+02	7.75E+00	3.88E+00	5.27E-03	2.63E-03
Cs-138	4.07E-01	7.43E+01	3.64E+00	1.82E+00	2.47E-03	1.24E-03
Cs-139	3.76E-02	6.86E+00	3.36E-01	1.68E-01	2.29E-04	1.14E-04
Ba-137m	8.15E-01	1.49E+02	7.29E+00	3.64E+00	4.95E-03	2.48E-03
Ba-139	3.04E-02	5.55E+00	2.72E-01	1.36E-01	1.85E-04	9.24E-05
Br-83	2.80E-02	5.11E+00	2.50E-01	1.25E-01	1.70E-04	8.51E-05
Br-84	1.40E-02	2.56E+00	1.25E-01	6.26E-02	8.51E-05	4.25E-05
Rb-86	1.22E-02	2.23E+00	1.09E-01	5.45E-02	7.42E-05	3.71E-05
Rb-88	1.06E+00	1.94E+02	9.48E+00	4.74E+00	6.44E-03	3.22E-03
Rb-89	6.21E-02	1.13E+01	5.55E-01	2.78E-01	3.77E-04	1.89E-04
Co-58	1.38E-02	2.52E+00	1.23E-01	6.17E-02	8.39E-05	4.19E-05
Tc-99m	4.47E-01	8.16E+01	4.00E+00	2.00E+00	2.72E-03	1.36E-03
Tc-101	8.32E-03	1.52E+00	7.44E-02	3.72E-02	5.06E-05	2.53E-05
Tc-102	6.25E-03	1.14E+00	5.59E-02	2.79E-02	3.80E-05	1.90E-05
Te-131m	7.14E-03	1.30E+00	6.38E-02	3.19E-02	4.34E-05	2.17E-05
Te-131	4.89E-03	8.93E-01	4.37E-02	2.19E-02	2.97E-05	1.49E-05
Te-132	7.78E-02	1.42E+01	6.96E-01	3.48E-01	4.73E-04	2.36E-04

Table III-2 (Continued)
Isotopes Activities in RCS and SG

Isotopes	Technical Specification DE I-131 (μCi/gm)	Primary Activity (Ci)	Unaffected SG Liquid Activity (Ci)	Affected SG Liquid Activity (Ci)	Unaffected SG Steam Activity (Ci)	Affected SG Steam Activity (Ci)
Te-133m	6.14E-03	1.12E+00	5.49E-02	2.74E-02	3.73E-05	1.87E-05
Te-133	3.46E-03	6.32E-01	3.09E-02	1.55E-02	2.10E-05	1.05E-05
Te-134	1.10E-02	2.01E+00	9.83E-02	4.92E-02	6.69E-05	3.34E-05
Mo-99	1.06E+00	1.94E+02	9.48E+00	4.74E+00	6.44E-03	3.22E-03
Mo-101	8.66E-03	1.58E+00	7.74E-02	3.87E-02	5.26E-05	2.63E-05
Mo-102	6.25E-03	1.14E+00	5.59E-02	2.79E-02	3.80E-05	1.90E-05

Table III-3
Technical Specification Iodine Concentrations and Appearance Rates

Nuclide	1 μCi/gm DE I-131 Tech Spec Concentrations (μCi/gm)	10 μCi/gm DE I-131 Pre-Accident Iodine Spike Concentrations (μCi/gm)
I-131	7.45E-01	7.45
I-132	3.76E-01	3.76
I-133	1.23E+00	12.3
I-134	2.42E-01	2.42
I-135	7.90E-01	7.90

Table III-4
Concurrent Reactor Coolant Iodine Spike Activities for SGTR and MSLB

Nuclide	SGTR Concurrent Spike Spike of 335 (Ci/hr)	MSLB Concurrent Spike Spike of 500 (Ci/hr)
I-131	7.550E+03	1.127E+04
I-132	1.011E+04	1.508E+04
I-133	1.447E+04	2.160E+04
I-134	1.315E+04	1.962E+04
I-135	1.254E+04	1.872E+04

Table III-5
Control Room and Offsite Atmospheric Dispersion Factors (χ/Q)

Control Room Atmospheric Dispersion Factors (χ/Q)			
Release Point	**Receptor Point**	**Time Interval (Hours)**	**Atmospheric Dispersion Factors (*seconds/cubic meter*)**
PORV	Normal CR Intake	0-SI Signal[1]	7.71E-03
PORV	Emergency CR Intake	SI Signal -720 hours	3.79E-03
Offsite Atmospheric Dispersion Factors (χ/Q)			
Receptor Point		**Time Interval (Hours)**	**Atmospheric Dispersion Factors (*seconds/cubic meter*)**
EAB		0-720	1.76E-03
LPZ		0-8	2.01E-04
LPZ		8-24	1.22E-04
LPZ		24-96	4.18E-05
LPZ		96-720	8.94E-06

1. Control room isolation is at time 0.0 hr for the LOOP condition (SGTR and MSLB) and at 0.0687 hrs. for the No-LOOP (SGTR only) condition due to an SI signal.

Table III-6
SGTR & MSLB Key Parameter Values

Description	***Parameter***
Primary System Volume (RCS)	8902 ft^3
Unaffected Steam Generator Liquid Volume	4104 ft^3
Affected Steam Generator Liquid Volume	2052 ft^3
SGTR Unaffected Steam Generator Steam Volume	1 ft^3
SGTR Affected Steam Generator Steam Volume	3889 ft^3
MSLB Steam Generator Steam Volume	1 ft^3
Turbine Building Volume (MSLB only)	6E6 ft^3
Control Room Volume	223,000 ft^3
Control Room Normal Ventilation (pre-isolation)(1)	3000 cfm
Control Room Normal Ventilation filter efficiency	NA
Control Room Emergency Ventilation(2) (filtered outside air)	1000 cfm
Control Room Emergency Ventilation filter efficiency	90% elemental iodine 70% organic iodine 99% particulates
Control Room Unfiltered Inleakage (post-isolation)	10 or 500 cfm
Measured Unfiltered Inleakage (Reference III-5)	147 ± 6 cfm
Total Primary-to-secondary leakage (accident induced)	1 gpm
RCS Full Power Temperature Avg.	574.4°F
RCS Pressure	2250 psia
RCS Density	45.22 lbm/ft^3 (0.724 gm/cc)
RCS Mass	1.826E+08 gm
SG liquid density	48.05 lbm/ft^3 (0.770 gm/cc)
SG liquid mass	4.470E+07 gm/SG
SG Steam density	1.723 lbm/ft^3 (0.028 gm/cc)
SG Steam Mass	3.039E+06 gm/SG
SG liquid Technical Specification limit on activity	0.1 µCi/gm DE I-131

Table III-6 (Continued)
SGTR & MSLB Key Parameter Values

Description	Parameter
Dilution Ratio (leak rate/steam flow rate)	3.15E-05
SG Iodine partitioning or Moisture Carryover	0.01
Breathing Rates and Occupancy Factors	Per RG 1.183
Dose Conversion Factors	FGR 11 and 12

1. Only applicable to the No-LOOP SGTR. Control room isolation is at time 0.0 hr for the LOOP condition (SGTR and MSLB) and at 0.0687 hrs. for the No-LOOP (SGTR only) condition due to an SI signal.
2. Manual alignment at time = 1 hour.

III.2.B.1.2 SGTR Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the affected steam generator by the break flow. A fraction of the break flow is assumed to flash to steam in the affected generator and to pass directly into the steam space of the affected generator with no credit taken for scrubbing by the steam generator liquid. The radionuclides initially in the steam space and those entering the steam space as the result of flashing pass directly to the environment through the Steam Generator PORVs or safety valves. The remainder of the break flow enters the steam generator liquid.

Releases of radionuclides initially in the steam generator liquid and those entering the steam generator liquid from the unflashed break flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are released from the steam generator liquid to the environment along with the steam flow. (Moisture carryover is not actually modeled but is instead addressed by application of the partitioning factor.) All noble gases are released from the primary system to the environment without reduction or mitigation. Releases were assumed to continue from the affected generator for 30 minutes until the affected generator was isolated. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is assumed to be transported to the unaffected generators by 1 gpm of primary-to-secondary leakage specified in the Technical Specifications. All radionuclides in the primary coolant leaking into the unaffected generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodine and particulates are assumed to pass directly to the environment. Radionuclides initially in the steam space are modeled to pass quickly to the environment. Again, all noble gases that are released from the primary system to the unaffected generator are released to the environment without reduction or mitigation. Releases were assumed to continue from the unaffected generator for a period of 8 hours until the primary system had cooled sufficiently to allow a switch over to the residual heat removal system.

III.2.B.1.3 SGTR Atmospheric Dispersion Factors (χ/Q)

The control room and the low population zone (LPZ) χ/Q values remain unchanged from the current license basis analysis (Reference III-6). Revised χ/Q values at the exclusion area boundary (EAB) which have been approved by the NRC (Reference III-4) were used. The control room, EAB, and LPZ χ/Q values used in the SGTR analysis are listed in Table III-5.

III.2.B.1.4 SGTR Key Analysis Assumptions and Inputs - Affected SG

The primary and secondary volumes along with the primary and secondary water and steam properties used in the analyses are provided in Table III-6. The steam generator flow rates used in the revised analyses are presented in Table III-7 for both the LOOP and No-LOOP cases. These flow rates have been updated to include new steam generator PORV relief flow as a function of steam pressure used to assess the flow capacity of the PORVs following the tube rupture as predicted by the accident analyses.

The LOCADOSE code was used to analyze cases for the pre-accident and concurrent iodine spikes with LOOP and with No-LOOP conditions. The limiting LOOP and No-LOOP cases for control room dose assume control room unfiltered intake airflow of 500 cfm and 10 cfm, respectively.

Table III-7
SGTR Flow Rates

(All flow rates are in cubic feet per minute)				
From Primary Coolant to Unaffected SG Liquid				0.1337 cfm (1 gpm)
From Unaffected Steam Generator Steam to Environment				1 cfm
LOOP				
Time (hr)	**RCS to Affected SG Liquid**	**RCS to Affected SG Steam**	**Affected SG Liquid to Steam(1)**	**Affected SG Steam to Environment**
0–0.0222	105	15.2	1330	0
0.0222–0.0625	104	5.48	168	4673
0.0625–0.5	85.2	6.61	96.9	2701

Table III-7 (Continued)
SGTR Flow Rates

Time (hr)	**Unaffected SG Liquid to Environment(1)**			
0–0.0228	0			
0.0228–0.0625	179			
0.0625–0.1186	66.0			
0.1186–0.5	0			
0.5–2	87			
2–8	33			
No-LOOP				
Time (hr)	**RCS to Affected SG Liquid**	**RCS to Affected SG Steam**	**Affected SG Liquid to Steam(1)**	**Affected SG Steam to Environment**
0–0.0731	91.4	9.83	1330	0
0.0731–0.1014	78.8	6.39	182	5088
0.1014–0.5	72.2	0.84	127	3541
Time (hr)	**Unaffected SG Liquid to Environment(1)**			
0–0.0947	0			
0.0947–0.1014	746			
0.1014–0.1497	173			
0.1497–0.5	0			
0.5–2	97			
2–8	49			

1. Partitioning and Moisture Carryover are modeled in the iodine and particulate releases by decreasing these flow rates by a factor of 100.

III.2.B.1.5 SGTR Analysis Results

The results of the SGTR MUR analysis for the Concurrent and Pre-Accident Iodine Spike are presented in Table III-8 along with the applicable UFSAR values. This table provides the dose consequences associated with the incremental changes that support this submittal. These changes include application of the approved EAB χ/Q, new MUR RCS source term, and PORV flow rates.

Table III-8
SGTR Dose Consequences[1]

Concurrent Iodine Spike - LOOP						
	Current UFSAR (Rem TEDE)	**Revised DB with Approved EAB χ/Q (Rem TEDE)**	**MUR Analysis with new RCS Source Term (Rem TEDE)**	**Increase due to MUR[4] %**	**Proposed MUR Dose Consequences including PORV flow increase (Rem TEDE)**	**Acceptance Criteria[3] (Rem TEDE)**
Control Room[2]	0.7	0.7	0.8	17%	1.3	5
EAB	2.2	0.9	1.0	15%	1.7	2.5
LPZ	0.2	0.2	0.2	17%	0.2	2.5
Pre-accident Iodine Spike - No-LOOP						
	Current UFSAR (Rem TEDE)	**Revised DB with Approved EAB χ/Q (Rem TEDE)**	**MUR Analysis with new RCS Source Term (Rem TEDE)**	**Increase due to MUR[4] %**	**Proposed MUR Dose Consequences including PORV flow increase (Rem TEDE)**	**Acceptance Criteria[3] (Rem TEDE)**
Control Room[2]	0.9	0.9	1.2	28%	4.3	5
EAB	1.7	0.7	0.8	20%	1.2	25
LPZ	0.1	0.1	0.1	24%	0.2	25

1. All dose values have been rounded up to one decimal place.
2. Control room unfiltered inleakage for the pre-accident iodine spike is 10 cfm and for the concurrent iodine spike is 500 cfm. The selection of 10 or 500 cfm of unfiltered inleakage was based on higher dose consequences.
3. RG 1.183 and 10 CFR 50.67
4. The increase is primarily due to an increase in primary and secondary Cs predicted in the updated RCS source term. The percentage change is based on actual calculated doses prior to rounding to the next highest 0.1 Rem TEDE.

III.2.B.2 Rupture of a Main Steam Pipe - UFSAR 14.3.2

The main steam line break (MSLB) accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine. The break is assumed to occur in the turbine building. The affected steam generator releases steam for 30 minutes, at which time it is isolated. Also, it is expected that the generator will dry out in 30 minutes. Loss of off-site power is assumed. As a result, the condenser is lost and cool-down of the primary system is through the release of steam from the unaffected generators. The release from the unaffected generators continues for 8 hours through the PORVs. To maximize both the control room and the off-site doses, air exhaust from the turbine building is modeled in two separate ways. Since the emergency intake for the control room takes suction from the turbine building, slow air exhaust from the turbine building is modeled (0.2 vol./hr.) to maximize the control room dose. To maximize the offsite dose, rapid air exhaust is modeled (12 vol./hr.). In accordance with RG 1.183, Appendix E, two independent cases are evaluated. First case assumes a pre-accident iodine spike above the value allowed for normal operation, while the second case assumes a concurrent iodine spike.

III.2.B.2.1 MSLB Source Term Definition

As with the SGTR accident, the analysis of the MSLB accident indicates that no fuel rod failures occur as a result of the transient. Thus, radioactive material released is the result of radionuclide concentrations initially present in primary liquid, secondary liquid, secondary steam, and any releases from fuel rods that failed before the transient. The Main Steam Line Break analysis uses the SGTR analysis source term, which is discussed in Section III.2.B.1.1. The only exception is that the MSLB accident assumes a concurrent accident iodine spike 500 times the release rate corresponding to the Technical Specification limit for normal operation (1 μCi/gm DE I-131) for a period of 8 hours, consistent with RG 1.183. The concurrent iodine spike appearance rates used in the MSLB accident analysis are shown in Table III-4. As mentioned in Section III.2.B.1.1, the radionuclide inventories in the updated RCS source term include a significant increase in the primary and secondary quantities of Cs.

The TEDE dose conversion factors used to calculate dose for the MSLB accident are built into the LOCADOSE library, which are consistent with Federal Guidance Reports 11 and 12 for the isotopes required by Regulatory Guide 1.183.

III.2.B.2.2 MSLB Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the steam generators by the accident induced leak rate of 1 gpm specified in the Technical Specifications (TS 6.4.Q.2.b). The maximum amount of accident induced primary-to-secondary leakage assumed to any one steam generator is 500 gallon per day. This leakage (500 gpd) was assigned to the affected generator. The remainder of the 1 gpm accident induced primary-to-secondary leakage was assigned to the two unaffected generators (modeled as one generator).

For the affected generator, all of the leakage flow is assumed to flash to steam and to pass directly into the turbine building with no credit taken for scrubbing by the steam generator liquid. The radionuclides initially in the steam generator liquid and steam pass directly to the turbine building through the broken steam line. From the turbine building it passes to the control room and to the environment. Releases were assumed to continue from the affected generator for 30 minutes until the affected generator was isolated. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

All radionuclides in the primary coolant leaking (940 gpd) into the unaffected generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodine and particulates are assumed to pass directly to the environment through the steam generator PORVs. (Moisture carryover is not actually modeled but is instead bounded by application of the partitioning factor.) Radionuclides initially in the steam space are modeled as passing quickly to the environment through the PORVs. All noble gases that are released from the primary system to the unaffected generator are released to the environment through the PORVs without reduction or mitigation. Releases were assumed to continue from the unaffected generator for a period of 8 hours until the primary system had cooled sufficiently to allow a switch-over to the residual heat removal system.

In the MSLB analysis, the rapid release of the initial activity in the Affected and Unaffected SG steam was accomplished by modeling a 1 ft^3 SG steam volume with a 1 cfm release flow rate. The release flow is split between the Turbine Building volume (1/3) and the environment (2/3).

III.2.B.2.3 MSLB Atmospheric Dispersion Factors

The control room, EAB, and LPZ values used in the MSLB analysis are the same as those used in the SGTR analysis, discussed in Section III.2.B.1.3 and presented in Table III-5.

III.2.B.2.4 MSLB Key Analysis Assumptions and Inputs

The primary and secondary volumes along with the primary and secondary water and steam properties used in the MSLB analyses are the same as those used in the SGTR analyses, provided in Table III-6. The flow rates used in the MSLB analyses are presented in Table III-9. Flow rates have been updated to include new steam generator PORV relief flow as a function of steam pressure required to achieve the cooldown capability imposed by the accident analyses.

The limiting case for the control room dose assumes a control room unfiltered inleakage airflow of 500 cfm. The LOCADOSE code was used to analyze cases for the pre-accident and concurrent iodine spikes with turbine building air exchange rates of 0.2 volumes/hour and 12 volumes/hour to determine maximum dose consequence conditions for the control room and EAB.

Table III-9
MSLB Release Flow Rates

(All flow rates are in cubic feet per minute)				
From Primary Coolant to Unaffected SG Liquid				0.0872 cfm
From Primary Coolant to Affected SG Liquid				0.0464 cfm
From Steam Generator Steam to Environment				0.666 cfm
From Steam Generator Steam to Turbine Building				0.334 cfm
Turbine Building Exhaust Flow Without Power – 0.2 volumes/hour				
Time	**Unaffected SG Liquid to Environment(1)**	**Affected SG Liquid to Turbine Building**	**Turbine Building Steam to Environment**	**Turbine Building Air to Environment**
0–41 sec	1669	1.632E+03	2.396E+06	2.000E+04
41–181 sec	0	3.818E+03	1.132E+06	2.000E+04
181–1800 sec	0	2.511E+03	4.096E+05	2.000E+04
0.5–2.0 hour	74	0	0	2.000E+04
2.0–8.0 hour	45	0	0	2.000E+04
Turbine Building Exhaust Flow With Power – 12 volumes/hour				
Time	**Unaffected SG Liquid to Environment(1)**	**Affected SG Liquid to Turbine Building**	**Turbine Building Steam to Environment**	**Turbine Building Air to Environment**
0–41 sec	1669	1.632E+03	2.396E+06	1.2000E+06
41–181 sec	0	3.818E+03	1.132E+06	1.2000E+06
181–1800 sec	0	2.511E+03	4.096E+05	1.2000E+06
0.5–2.0 hour	74	0	0	1.2000E+06
2.0–8.0 hour	45	0	0	1.2000E+06
1. Partitioning and Moisture Carryover are modeled in the iodine and particulate releases by decreasing these flow rates by a factor of 100.				

III.2.B.2.5 MSLB Analysis Results

The results of the MSLB MUR analysis for the Concurrent and Pre-Accident Iodine Spike cases are presented in Table III-10 along with the applicable UFSAR values. This table provides the dose consequences associated with the incremental changes that support this submittal. These changes include application of the approved EAB χ/Q, new MUR RCS source term, and PORV flow rates.

Table III-10

MSLB Dose Consequences(1)

Concurrent Iodine Spike - LOOP						
	Current UFSAR (Rem TEDE)	**Revised DB with Approved EAB χ/Q (Rem TEDE)**	**MUR Analysis with new RCS Source Term (Rem TEDE)**	**Increase due to MUR(4) %**	**Proposed MUR Dose Consequences including PORV flow increase (Rem TEDE)**	**Acceptance Criteria(3) (Rem TEDE)**
Control Room(2)	0.7	0.7	1.5	137%	1.6	5
EAB	0.4	0.2	0.4	195%	0.5	2.5
LPZ	0.1	0.1	0.1	123%	0.1	2.5
Pre-accident Iodine Spike No-LOOP						
	Current UFSAR (TEDE Rem)	**Revised DB with Approved EAB χ/Q (Rem TEDE)**	**MUR Analysis with new RCS Source Term (Rem TEDE)**	**Increase due to MUR(4) %**	**Proposed MUR Dose Consequences including PORV flow increase (Rem TEDE)**	**Acceptance Criteria(3) (Rem TEDE)**
Control Room(2)	0.5	0.5	1.4	187%	1.4	5
EAB	0.4	0.2	0.4	209%	0.4	25
LPZ	0.1	0.1	0.1	200%	0.1	25

1. All dose values have been rounded up to one decimal place.
2. Based on control room unfiltered inleakage of 500 cfm.
3. RG 1.183 and 10 CFR 50.67.
4. The increase is primarily due to an increase in primary and secondary Cs predicted in the updated RCS source term. The MSLB results are very sensitive to the increase in Cs inventory in the SG liquid and primary-to-secondary leakage because no partitioning occurs in the faulted SG. The percentage change is based on actual calculated doses prior to rounding to the next highest 0.1 Rem TEDE.

III REFERENCES

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III-2 Letter from J. P. O'Hanlon (Dominion) to U.S. Nuclear Regulatory Commission, *Surry Power Station Units 1 and 2, Proposed Technical Specification Changes to Accommodate Core Uprating,* Serial No. 94-509, August 30, 1994.

III-3 Letter from B. C. Buckley (NRC) to J. P. O'Hanlon (Dominion) *Surry Units 1 and 2, Issuance of Amendments Re: Uprated Core Power (Serial No. 94-509) (TAC Nos. M90364 and M90365),* Serial No. 95-405, August 3, 1995.

III-4 Letter from Stephen Monarque (NRC) to David A. Christian (Dominion) *Surry Power Station, Units 1 and 2, Issuance of Amendments Regarding The Redefinition of The Exclusion Area Boundary, (TAC Nos. MC8315 and MC83165),* Serial No. 06-701, August 10, 2006.

III-5 Letter from D. A. Christian (Dominion) to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2; Generic Letter 2003-01 - Control Room Habitability Control Room Testing and Technical Information Submittal,* Serial No. 03-373C, April 22, 2004.

III-6 Letter Serial No. 02-170, *Surry Units 1 and 2 - Issuance of Amendments Re: Alternative Source Term (TAC Nos. MA8649 and MA8650),* March 8, 2002.

III-7 Letter Serial No. 89-381A, *Surry Power Station, Units 1 and 2, Control Room Dose Calculations/Habitability Assessment Proposed Operating License Amendment,* October 26, 1989.

IV. MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

1. A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.

   A. This discussion should address the following components:

      i. reactor vessel, nozzles and supports
      ii. reactor core support structures and vessel internals
      iii. control rod drive mechanisms
      iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles
      v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, containment systems)
      vi. SG tubes, secondary side internal support structures, shell, nozzles
      vii. RCPs
      viii. pressurizer shell, nozzles, surge line
      ix. safety-related valves

   B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:

      i. stresses
      ii. cumulative usage factors (fatigue)
      iii. flow induced vibration
      iv. changes in temperature (pre- and post-uprate)
      v. changes in pressure (pre- and post-uprate)
      vi. changes in flow rates (pre- and post-uprate)
      vii. high energy line break locations
      viii. jet impingement and thrust forces

   C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel integrity with respect to:

      i. pressurized thermal shock calculations
      ii. fluence evaluation
      iii. heatup and cooldown pressure-temperature limit curves
      iv. low temperature overpressure protection
      v. upper shelf energy
      vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the *associated analyses, and any changes to the code of record.*

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs, and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube cycle fatigue is consistent with NRC Bulletin 88-02, *Rapidly Propagating Fatigue Cracks in Steam Generator Tubes*, February 5, 1988.

RESPONSE TO IV - MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

IV.1.A.i Reactor Vessel

The reactor vessel stress and fatigue usage factors were evaluated at the uprated operating conditions. The evaluation assessed the effects of the revised operating parameters on the most limiting locations. The SPS reactor vessels were originally analyzed with a normal operating inlet temperature of 543.0°F and a normal operating outlet temperature of 606.0°F. In support of the 1995 stretch power uprate, the analyzed normal operating temperatures were modified to a vessel inlet temperature of 540.4°F and vessel outlet temperature of 605.6°F, to agree with the design parameters. The minimum vessel inlet temperature for normal operation decreased from 540.4°F to 536.7°F with the MUR power uprate. The maximum vessel outlet temperature for normal operation increased from 605.6°F to 609.1°F with the MUR power uprate. Neither the minimum vessel inlet temperature nor the maximum vessel outlet temperature were bounded by current analysis and required further evaluation.

The reactor vessel main closure flange assembly, CRDM housings, and outlet nozzles were evaluated for the effect of the increase in maximum vessel outlet temperature from 605.6°F to 609.1°F. The remaining reactor vessel regions are assumed to be in contact with vessel inlet water during normal operation. These regions were evaluated for the effect of the decrease in the minimum vessel inlet temperature from 542.9°F to 536.7°F. The maximum ranges of primary-plus-secondary stress intensity reported for the closure head flange, vessel flange, closure studs, CRDM housings, and outlet nozzles were evaluated. None of the maximum ranges of stress intensity exceeded the $3S_m$ limit, and the maximum CUF continues to remain below the 1.0 acceptance criterion.

The code of record is listed in Section IV.1.D and remains unchanged. The reactor vessel meets the stress and fatigue analysis requirements of ASME B&PV Code, Section III, for plant operation at the uprated power conditions.

IV.1.A.ii Reactor Vessel Internals

The revised design conditions were evaluated for impact on the existing reactor vessel internals design basis analyses. The reactor vessel internals evaluations concluded that these components continue to meet their design criteria at the MUR power uprate conditions. The basis for those conclusions is specified below.

IV.1.A.ii.a Core Bypass Flow

The design core bypass flow limit is 6.0% of the total reactor vessel flow. This core bypass flow limit remains unchanged and valid for power uprate conditions. The MUR power uprate has an insignificant effect on the core bypass flow; core bypass flow remains below the 6.0% limit.

IV.1.A.ii.b Rod Control Cluster Assembly Drop Time

An analysis was performed to demonstrate that the RCCA drop time is still within the current Technical Specification value of 2.4 seconds for the revised design conditions. The analysis indicated that the revised design conditions will have an insignificant impact on the RCCA drop time, and the estimated rod drop time will remain less than 2.4 seconds.

IV.1.A.ii.c Hydraulic Lift Forces and Pressure Losses

An evaluation was performed to determine the hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly remains seated and stable for the applicable design conditions. The results indicate that the downward force remains essentially unchanged, and the reactor internals would remain seated and stable at the MUR power uprate conditions.

IV.1.A.ii.d Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of the fuel rod caused by a high-velocity water jet. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core. The baffle jetting phenomenon could lead to fuel cladding damage. There is no significant change to the pressure differential across the baffle plate, baffle gap width, and fuel assembly modal response due to the power uprating. Therefore, the baffle joint momentum flux would not change as a result of the MUR power uprate.

IV.1.A.ii.e Part-Length CRDM Removal

Evaluations were performed for the removal of the five part-length CRDM lead screws, with a flow restrictor device installed in one part-length CRDM location for only Surry Unit 1. These modifications were performed previously as part of the 2003 reactor vessel head replacements at Units 1 and 2, respectively. The potential effects on the reactor vessel and internals fluid systems thermal-hydraulic characteristics were assessed using comparative calculations made with the THRIVE computer code. Assessment results indicated that the part-length CRDM head penetration modification did not significantly affect pressure drop, lift forces and reactor internals core bypass flows in regards to thermal hydraulic performance of the reactor pressure vessel system.

IV.1.A.ii.f Mechanical Evaluation

The power uprated conditions do not affect the current design bases for seismic and LOCA loads. The flow induced vibration stress levels on the core barrel assembly and upper internals are low and below the material high-cycle fatigue endurance limit. Therefore, the MUR uprated conditions do not affect the structural margin for flow-induced vibration.

IV.1.A.ii.g Structural Evaluation

Evaluations were performed to demonstrate that the structural integrity of reactor internal components is not adversely affected by the MUR power uprate. For reactor internal components except the lower core plate and the upper core plate, the stresses and cumulative usage factor of the previous analyses remain bounding at power uprate conditions.

IV.1.A.ii.g.1 Lower Core Plate Structural Analysis

The lower core plate is subjected to the effects of heat generation rates, due to its proximity to the core. Structural evaluations were performed to demonstrate that the lower core plate structural integrity was not adversely affected by the revised design conditions. The lower core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, is acceptable. The lower core plate is structurally adequate for the MUR power uprate conditions.

IV.1.A.ii.g.2 Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel with installed baffle plates. Bolting connects former plates to the baffle and core barrel. This bolting restrains baffle plate motion. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA, and seismic loads, and secondary loads consisting of preload and thermal loads resulting from RCS temperatures and gamma heating rates.

An evaluation of the baffle former bolt maximum displacement was performed at MUR power uprate conditions. This displacement is caused by the temperature difference between the baffle and barrel regions, which is influenced by the power in the fuel assemblies adjacent to the baffle plates. The original analysis assumed that fresh fuel assemblies were loaded adjacent to the baffle. Power on the peripheral fuel assemblies is less than the initial power distribution, because only irradiated assemblies are loaded in the peripheral core locations. The core power distribution (lower power levels of peripheral fuel assemblies) offsets the increased loads due to gamma heating rates, resulting in a temperature difference less than the previous analysis of record. Therefore, the existing baffle-barrel region thermal and structural analysis results remain bounding for the MUR revised design conditions.

IV.1.A.ii.g.3 Upper Core Plate Structural Analysis

The maximum stress contributor in the upper core plate is the membrane stress resulting from the average temperature difference between the center portion of the upper core plate and the rim. The increased stress from increased gamma heating was determined as a function of heat generation rate increment. The fluid temperature effect resulting from the power uprate was small. The evaluation results indicate that the upper core plate structural integrity is maintained at power uprate conditions. The upper core plate maximum primary plus secondary stress intensity and cumulative usage factor, including the effects of increased heat generation rates, are acceptable. The upper core plate is structurally adequate for the MUR power uprate conditions.

IV.1.A.iii Control Rod Drive Mechanism

The CRDMs use electro-magnetic coils to position the RCCA within the reactor core. The updated design conditions (design parameters and NSSS design transients) were reviewed for impact on the existing CRDM design basis analyses. CRDMs are designed for T_{hot} temperatures and RCS pressures. These parameters were reviewed in the CRDM evaluation. The maximum T_{hot} from the uprated design parameters for any case is 609.1°F for Surry Unit 1, which is equal to the maximum T_{hot} used in the Surry Unit 1 analysis of record. The maximum T_{hot} from the uprated design parameters for any case is 609.1°F for Surry Unit 2, which is less than the 610°F maximum T_{hot} used in the Surry Unit 2 analysis of record. No changes in RCS design or operating pressure were made as part of the power uprate. Since the Surry Unit 1 design parameters and NSSS design transients are unchanged from the analysis of record, the Surry Unit 1 CRDM stresses are not impacted by the power uprate and remain acceptable. The Surry Unit 2 design parameters and NSSS design transients are bounded by the analysis of record. Therefore, the Surry Unit 2 CRDM stresses are acceptable for the power uprate. The fatigue analyses for both Surry Units 1 and 2 remain acceptable for the power uprate. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.iv Reactor Coolant Piping and Supports

The revised design conditions for the MUR uprate were evaluated for impact on the existing design basis analyses for the reactor coolant loop piping, including the loop bypass line and the pressurizer surge line, primary equipment nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge), primary equipment supports (reactor pressure vessel nozzle supports, SG supports, and RCP supports), reactor coolant loop branch nozzles (accumulator and charging line), and Class 1 auxiliary piping systems attached to the reactor coolant loop. There are no significant changes to the reactor coolant loop thermal analysis, LOCA analysis and main steam line break analysis due to the MUR uprate conditions. SPS reactor coolant loop piping and pressurizer surge line piping are designed to USAS B31.1, which does not require fatigue evaluations. There is no significant impact due to NSSS design transients on the RCS piping.

Therefore, the MUR power uprate design parameters have insignificant impact on the reactor coolant loop piping analyses and evaluations including: reactor coolant loop piping stresses, primary equipment nozzles, primary equipment supports, Class 1 auxiliary piping lines attached to the reactor coolant loops and the Class 1 auxiliary line branch nozzles attached to the reactor coolant loops. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.v Balance-of-Plant Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems and Containment Systems)

BOP piping includes NSSS interface systems, safety-related cooling water systems, and containment systems. The MUR uprate operating conditions for the BOP piping were reviewed for impact on the existing piping and supports design basis analyses.

Change factors were determined, as required, to evaluate and compare the changes in operating conditions. Thermal, pressure, and flow rate change factors were based on the following ratios.

- The thermal change factor was based on the ratio of power uprate to pre-uprate operating temperature $(T_{uprate} - 70°F)/(T_{pre\text{-}uprate} - 70°F)$
- The pressure change factor was determined by the ratio of $(P_{uprate}/P_{pre\text{-}uprate})$
- The flow rate change factor was determined by the ratio of $(Flow_{uprate}/Flow_{pre\text{-}uprate})$

These thermal, pressure and flow rate change factors were used in determining piping systems acceptability for power uprate conditions. When the change factors are ≤1.0 (the analyzed condition envelops or equals the power uprate condition), the piping system was considered acceptable for power uprate conditions. When the change factors are >1.0, an evaluation was performed to address the specific temperature, pressure and/or flow rate increase to document piping system acceptability.

The following Units 1 and 2 BOP and NSSS interface piping systems were evaluated for MUR uprate conditions:

<u>BOP Piping Systems</u>
Auxiliary Feedwater System
Fuel Pool Cooling and Cleanup System
Containment Spray and Recirculation Spray Systems
Circulating Water System
Component Cooling Water System
Main Steam and Steam Dump System
Extraction Steam System
Condensate System
Feedwater System
Heater Drains System
Service Water System
Bearing Cooling Water System
Auxiliary Steam System
Chilled Water System
Gaseous Waste System
Liquid Waste System
Service and Instrument Air System

<u>NSSS Interface Piping Systems</u>
Chemical and Volume Control System
Residual Heat Removal System
Safety Injection System
Pressurizer Spray System
Pressurizer Safety Relief Valve and Pressurizer Power Operated Relief Valve System
Steam Generator Blowdown System

The design basis requirements for BOP systems were reviewed for changes in the temperature, pressure and flow rate effects resulting from the MUR power uprate conditions. The changes are acceptable.

IV.1.A.vi Steam Generator

The original Unit 1 and 2 Model 51 SGs were replaced in 1981 and 1980, respectively. The Model 51F replacement SGs are a blend of a new tube bundle, lower shell, primary channel head region, and primary moisture separators and feedrings, with the original upper shell and secondary moisture separators (Model 51 steam drum). The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.vi.1 Steam Generator Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation focused on changes to secondary side operating characteristics at MUR power uprate conditions. SG secondary side performance characteristics such as steam pressure and flow, circulation ratio, bundle mixture flow, heat flux, secondary side pressure drop, moisture carryover, hydrodynamic stability, secondary side mass and others are affected by increases in power level. Secondary side performance characteristics were calculated using the SG performance code GENF (secondary side characteristics except DNB). GENF code analyses were performed for the design parameter cases. A separate analysis was performed using the 3-D flow field analysis code ATHOS (DNB parameters) to determine the detailed flow parameters throughout the tube bundle. The thermal-hydraulic evaluation concluded that SPS SG thermal-hydraulic operating characteristics remain acceptable for the MUR power uprate.

IV.1.A.vi.2 Steam Generator Structural Integrity

The structural evaluation focused on the critical SG components as determined by the design basis analyses stress ratios and fatigue usages.

For primary side components (including the divider plate, tubesheet and shell junctions, tube-to-tubesheet weld, and tubes), the applicable scale factors were the ratios of the baseline condition primary-to-secondary side differential pressures to the uprated conditions differential pressures. The scale factor was applied conservatively to both the thermal and pressure stresses. For the secondary side components (including the feedwater nozzle and secondary manway bolts), the decrease in secondary side pressure at uprated conditions was the basis for determining the applicable scale factors. The stress increase resulting from the steam pressure reduction was calculated. The additional stress was then used in calculating the resulting fatigue usage changes for operation at MUR power uprate conditions. The scale factors were applied to the stresses listed in the reference stress reports. The scaled stresses were also considered in determining the stress ranges involving transients that originate from, or lead to, full power.

An analysis was performed to determine if the ASME B&PV Code limits on design primary-to-secondary ΔP are exceeded for any applicable transient at power uprate conditions. The analysis determined that the maximum primary to secondary side differential pressures during normal operating transients are 1549 psid and 1593 psid for high T_{avg} and low T_{avg} temperatures respectively. The maximum primary to secondary side differential pressure during upset condition transients is 1469 psid and 1513 psid for high T_{avg} and low T_{avg} temperatures respectively. These values are below the applicable design pressure limits of 1600 psid and 1760 psid for normal and upset conditions respectively. Therefore, the ASME B&PV Code design pressure requirements are satisfied.

The primary-plus-secondary stress range for primary side and secondary side components was evaluated. The maximum range of primary-plus-secondary stress was compared with the corresponding $3S_m$ limit of the ASME B&PV Code. The analyzed components meet the ASME B&PV Code limits. Cumulative usage factors for affected components remain below 1.0.

IV.1.A.vi.3 Steam Generator Tube Bundle Integrity, Flow Induced Vibration and Wear

Tube Integrity

The SPS Model 51F replacement SGs contain thermally treated Alloy 600 tubing and ASME SA-240 tube support plates with broached quatrefoil holes. The quatrefoil tube hole configuration results in reduced potential for contaminant concentration at tube support plate intersections by reducing the crevice area. The first eight tube rows were heat treated after bending to relieve stresses. Hydraulic tube expansion in the tubesheet region results in reduced residual stresses compared to mechanical roll expansion and a more uniform expansion compared to explosively expanded tubes. Thermally treated Alloy 600 provides improved resistance to stress corrosion cracking as compared with tube material in the original SGs. The replacement SGs have exhibited little corrosion-related tube degradation after seventeen cycles in Unit 1 and eighteen cycles in Unit 2. Actual tube plugging levels (from corrosion and other causes) are Unit 1 - 0.86% (86 tubes) and Unit 2 - 0.94% (94 tubes). In both units, indications of primary side tube corrosion have been identified near the tube end; this condition is acceptable for service in accordance with the current Technical Specification requirements. The only other recent indication of corrosion in either unit was identified during the Spring 2009 Unit 1 outage; one tube, identified as having high residual stress, exhibited primary water stress corrosion cracking at the top of the tubesheet. In the mid-1990s several indications characterized as pitting were identified in both units; none of these indications has exhibited growth since chemical cleaning was performed in the mid-1990s, and initiation of new pits is unlikely using modern secondary chemistry management.

Service-induced mechanical tube degradation mechanisms have been identified in both units. These include foreign object-related wear, anti-vibration bar wear, and tube support plate wear. These degradation mechanisms have typically progressed slowly and have caused only a modest number of tube repairs.

Condition monitoring and operational assessments performed to date confirm that the SG tube integrity performance criteria were met during operation and are expected to continue to be met during the operating period prior to the next scheduled examination in each unit.

Potential tube degradation mechanisms resulting from potential localized chemistry changes at the tube surfaces after the power uprate are outside diameter stress corrosion cracking and pitting. Based on laboratory and operating experience and current SPS operating and maintenance practices, the power uprate will not produce excessive degradation due to those mechanisms. On the basis of T_{hot} temperature increase alone, the mechanical wear processes are unlikely to be significantly changed. The increased RCS temperature effects on primary water stress corrosion cracking are expected to be small. SG chemistry effects on tube corrosion after power uprating are insignificant.

Flow Induced Vibration and Wear

SG tube wear (i.e., fretting) was evaluated based on current design basis analysis and consideration of SG secondary side thermal-hydraulic changes resulting from the MUR power uprate. SG tube wear due to fluid-elastic effects in the U-bend region and turbulence induced displacement effects in the straight leg tube region were considered.

The analysis results indicate an increase in fluidelastic stability of as much as 13%, with an increase in vibration amplitude due to turbulence and an increase in tube wear of as much as 29%. This results in a maximum stability ratio of 0.57, which is < 1.0 allowable and acceptable. The maximum turbulence induced amplitude is < 0.006 inch, which is less than half the distance between tubes and acceptable. The maximum post-uprate wear over 40 years is < 0.002 inch. This projected wear is an increase of approximately 50% from 0.0011 inch (pre-uprate). This amount of tube wear will not significantly affect tube integrity, and is acceptable.

Other items reviewed were tube stress and fatigue. Tube stress resulting from flow induced vibration concerns after the MUR power uprate is approximately 0.2 ksi. This stress level is below the ASME stress limits and the fatigue endurance limit. Therefore, tube stresses are acceptable at MUR power uprate conditions, the flow induced vibration induced loading fatigue usage factor is negligible, and fatigue degradation from flow induced vibration is not anticipated.

IV.1.A.vi.4 Steam Generator Steam Drum Evaluation

The Model 51F replacement SGs are a blend of a new tube bundle, lower shell, primary channel head region, primary moisture separators and feedrings, with the original upper shell and secondary moisture separator (Model 51 steam drum). Performance improvement modifications were made to the Model 51F SG steam drums in 1990 for Unit 1, and 1991 and 1995 for Unit 2. These physical modifications were made to reduce moisture carryover and address primary moisture separator degradation due to FAC.

FAC in the SG steam drum region depends on numerous factors, including material composition, fluid velocity and turbulence, and secondary side water chemistry. Operation at uprated plant conditions will increase feedwater flow rates in the SGs with the possibility of initiating or accelerating the FAC process within the steam drum regions. Feedwater ring degradation due to FAC has been observed through UT thickness measurements. However, even with the observed degradation, the SG feedwater ring thermal performance should be maintained within the originally specified design conditions during MUR power uprate operation. There is minimal concern from a FAC standpoint for the primary separators, because they were previously replaced with wear resistant Alloy 600. Dominion will continue to perform steam drum component inspections to determine if the increased feedwater flow rates have initiated or accelerated the FAC process.

IV.1.A.vi.5 Steam Generator Mechanical Repair Hardware

Mechanical repair hardware refers to components such as plugs, sleeves, and stabilizers that are installed in the SGs to address tube degradation.

Analysis results showed that mechanical plug designs satisfy applicable stress, fatigue and retention acceptance criteria for operation at MUR power uprate conditions. There are shop welded plugs containing Alloy 600 in the SPS replacement SGs. The NPT-80 field installed weld plug may be used in applications that cannot employ a mechanical plug. Evaluations determined that the shop welded plugs and the NPT-80 weld plugs remain qualified at power uprate conditions. Field machining SG tube ends is a possibility for modifications and tube repair (i.e., plugging, sleeving and tube end reopening). The analysis concluded that the revised stresses were within the ASME B&PV Code allowable values. The fatigue usage values, when adjusted for the power uprating, remained less than the 1.0 fatigue limit. Straight-leg sleeved cable stabilizers remain qualified for SPS.

Therefore, SG repair hardware continues to meet ASME B&PV Code limits for plant operation at MUR power uprate conditions.

IV.1.A.vi.6 Steam Generator Loose Parts/Foreign Objects

Foreign object search and retrieval operations during previous Surry refueling outages determined that four unretrievable objects are present in the Unit 1 SGs (as of the Spring 2009 refueling outage) and five in the Unit 2 SGs (as of the Fall 2009 refueling outage).

The previous loose part evaluations were reviewed to determine the power uprate effects on the object projected wear times. Although there was no indication of wear present on any tubes adjacent to the foreign object, the wear time analyses were performed assuming 20% initial tube wear on the limiting tube location. The SG secondary side conditions will change as a result of the MUR operating conditions; however, these changes do not affect the previous evaluation conclusions.

The analysis determined that the amount of time required for the limiting foreign object orientation to wear a tube down to a minimum allowable tube wall thickness under conservative secondary side conditions was greater than 3 years or 2 operational cycles.

Therefore, operation at the MUR power uprate conditions is acceptable with the existing SG foreign objects.

IV.1.A.vi.7 Regulatory Guide 1.121 Analysis

NRC Regulatory Guide 1.121 describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by inservice inspection must be repaired or removed from service. The acceptable degradation level is called the repair limit.

The Regulatory Guide 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. SG tubing structural limits were determined by analysis, for an assumed uniform thinning degradation mode in both the axial and circumferential directions. The allowable stress limits were established using the ASME B&PV Code, Section III, 1986, Code Case N-20-3 minimum strength properties. The limiting stresses during normal operation (Level A) and upset (Level B) service conditions are the primary membrane stresses due to the primary-to-secondary pressure differential across the tube wall. The postulated accident condition loads for the faulted (Level D) service condition are the LOCA+DBE (design basis earthquake) and steam line break+DBE.

The allowable tube repair limit, per Regulatory Guide 1.121, is established by adjusting the structural limit to take into account uncertainties in eddy current measurement, and an operational allowance for continued tube degradation until the next scheduled inspection. Analyses have been performed to establish the structural limit for the tube straight-leg (free span) region for degradation over an unlimited axial extent, and for degradation over a limited axial extent at the tube support plate and anti-vibration bar intersections. The existing tube repair limit is unaffected by the MUR power uprate and remains valid at uprate conditions.

IV.1.A.vii Reactor Coolant Pumps and Reactor Coolant Pump Motors

Updated RCS conditions were reviewed for impact on the existing RCP design basis analyses. The NSSS design parameters considered in the RCP evaluation are vessel inlet temperatures and RCS pressure. The reactor vessel inlet temperature at the RCP discharge is considered instead of the SG inlet temperature at the RCP inlet because the vessel inlet temperature is slightly higher due to pump heat. No changes in RCS design or operating pressure were made as part of the power uprate. The maximum vessel inlet temperature for any NSSS design parameters case is 542.9°F. This temperature is essentially the same as the previously evaluated vessel inlet temperature of 543°F. The power uprate conditions remain bounded by the original design parameters conditions. The NSSS transient conditions are bounded for each RCP pressure boundary component, with the exception of the weir plate. Additional calculations verified that the weir plate remains within the allowable ASME B&PV Code stress limits. A fatigue waiver per ASME B&PV Code Section NB-3222.4(d) was performed for the weir plate original qualification. This fatigue waiver remains applicable to the uprated conditions.

The RCP motor limiting design parameter is the horsepower loading at continuous hot and cold operation. The new worst-case RCP motor loads are 6150 horsepower for the hot loop condition and 7777 horsepower for the cold loop condition. These loadings are larger than the RCP motor nameplate ratings of 6000 horsepower for hot loop operation and 7500 horsepower for cold loop operation. The RCP motors were evaluated under the revised loading to determine acceptability. A previous evaluation was conducted for the same RCP motors at a hot loop load of 6317 horsepower, which is bounding for the MUR uprate worst case hot loop load of 6150 horsepower. A previous evaluation was conducted for the same RCP motors at a cold loop load of 8006 horsepower, which is bounding for the MUR uprate worst case cold loop load of 7777 horsepower. The temperature rises associated with the revised hot loop loading, cold loop loading, and starting conditions comply with the RCP motor specification requirements, and are acceptable. The thrust bearing loading changes at MUR uprate conditions were not significant. The thrust bearings are acceptable for the revised loads. Therefore, the RCP motors are acceptable for MUR power uprate conditions.

The updated RCS conditions are acceptable for the RCP with respect to ASME B&PV Code structural integrity. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.viii Pressurizer Structural Evaluation

The MUR operating conditions were reviewed for impact on the existing pressurizer design basis analysis. The limiting pressurizer conditions occur when the RCS pressure is high and the RCS T_{hot} and T_{cold} are low. No changes were made in RCS design or operating pressure as part of the power uprate. The minimum T_{hot} and T_{cold} values from the design parameter cases were used in the pressurizer evaluation. The T_{hot} change was minimal and bounded by the original design basis, no analyses were necessary for the lower shell and its key components. The change in T_{cold} warranted an analysis of key upper shell components (spray nozzle, safety and relief nozzle, and the upper shell). The upper shell fatigue usage decreased due to removing excess conservatism from the original evaluation.

The NSSS design transients did not change and were enveloped by the existing design transients. Pressure fluctuations during the uprate transients are the same as the pressures in the original evaluations. Therefore, the power uprate transients have no effect on the primary stress evaluations previously performed for each load category (Normal, Upset, Faulted, and Test).

The SPS pressurizer lower head was previously evaluated for insurge/outsurge transient effects related to both design transients and operational transients that were not considered in the original design. The revised design parameters were evaluated for their effect on the previous evaluation conclusions. The revised design transients caused minor changes to the previous fatigue results. The fatigue results remain within the allowable limits of the ASME Code.

Therefore, the pressurizer meets the stress/fatigue analysis requirements for plant operation at MUR power uprate conditions. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.ix Safety Related Valves

The revised design conditions were reviewed for impact on the existing safety-related valves design basis analyses. No changes in RCS design or operating pressure were made as part of the power uprate. The evaluations concluded that the temperature changes due to the power uprate are bounded by those used in the existing analyses. Safety-related valves were reviewed within the applicable system (Section VI) and program (Section VII.6.E) evaluations. None of the safety-related valves required a change to their design or operation as a result of the power uprate.

IV.1.A.x Loop Stop Isolation Valves

The updated design conditions were reviewed for impact on the existing loop stop isolation valve design basis analyses. No changes in RCS design or operating pressure were made as part of the power uprate. The loop stop isolation valves are located in each RCS hot leg and cold leg. Higher temperatures are more limiting for the design qualification, so the hot leg valves were chosen to bound both applications. The maximum T_{hot} from any design parameters case is 609.1°F. This value is below the loop stop isolation valve design temperature of 650°F and the T_{hot} of 611°F used in the design analysis. Thus, the increased hot leg temperature is bounded by the original loop stop isolation valve evaluations. The existing NSSS design transients used in the fatigue analysis bound the power uprate transients with the exception of the 10% step load change and loss of power. These transients were reanalyzed and the results were combined with other transients to determine the fatigue usage factor. The new fatigue usage factor is only slightly greater than the original usage factor and remains below the ASME B&PV Code allowable value of 1.0.

Therefore, the loop stop isolation valves are acceptable with respect to revised performance parameters and transients. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.B.i Stresses

The revised design conditions for the NSSS components and BOP piping (NSSS interface systems, safety-related cooling water systems and containment systems) were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and cumulative usage factors) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping and supports), IV.1.A.vi (steam generator), IV.1.A.vii (RCPs and motors), IV.1.A.viii (pressurizer), IV.1.A.ix (safety-related valves), and IV.1.A.x (loop stop isolation valves). No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes are within design limits. The evaluations reviewed maximum stress intensities/stress ranges, with comparison to stress allowables, cumulative usage factors (for Class 1), and other special stress limits. The MUR power uprate transient conditions are bounded by the design transient conditions.

IV.1.B.ii Cumulative Usage Factors

The revised design conditions for the NSSS components, piping, and interface systems were reviewed for impact on the existing design basis analyses. Structural evaluations (stress and cumulative usage factors) are discussed in Sections IV.1.A.i (reactor vessel), IV.1.A.ii (reactor vessel internals), IV.1.A.iii (control rod drive mechanism), IV.1.A.iv (reactor coolant piping), IV.1.A.vi (steam generator), IV.1.A.vii (RCPs and motors), IV.1.A.viii (pressurizer), IV.1.A.ix (safety-related valves), and IV.1.A.x (loop stop isolation valves).

For Class 1 components, the stress analyses considered the impact on fatigue life. The cumulative usage factors were determined to be acceptable (< 1.0) for a 60-year plant life for the components.

IV.1.B.iii Flow Induced Vibration

SG flow induced vibration is discussed in Section IV.1.A.vi.3.

IV.1.B.iv Temperature Effects

IV.1.B.iv.1 Changes in Temperature (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the SPS power uprate. The NSSS design temperature values are shown in Attachment 1 Table 4.0-2. Specific calculation outputs include T_{hot} and T_{cold}. There is an approximate 1.0°F increase in temperature across the core from current operating conditions due to the MUR power uprate.

Changes in main steam and feedwater system temperatures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.iv.2 Evaluation of Potential for Thermal Stratification

NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems,* addresses thermal stresses in piping attached to the RCS that cannot be isolated. The MUR power uprate temperature changes, when compared to current operation and evaluated using EPRI Material Reliability Program, MRP-146 (Reference IV-9), will not cause changes in the potential for cyclical thermal stratification, or in the predicted temperature profiles and cycling frequencies, that would require any different management approach to this issue from the existing Dominion programs. In addition, the RCS design flow rates are essentially the same as the power uprate values. Thus, the effects of swirl penetration will not change due to the power uprate.

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification,* addresses surge line thermal stratification. Surge line thermal stratification is driven by the temperature difference between the RCS hot leg and the pressurizer. The current hot leg operating temperature will be higher for the power uprate. Hence, the current analysis envelops the MUR condition.

IV.1.B.v Changes in Pressure (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the SPS power uprate. There will be no change in RCS operating pressure as a result of the MUR power uprate. The nominal operating pressure is 2250 psia as shown in Attachment 1 Table 4.0-2.

Changes in main steam and feedwater system pressures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vi Changes in Flow Rates (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for the SPS power uprate. The mechanical design RCS flow is shown in Attachment 1 Table 4.0-2 and remains unchanged for the MUR power uprate.

Changes in main steam and feedwater system flow rates are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.vii High Energy Line Break

IV.1.B.vii.1 High Energy Line Break Locations

A review was performed to determine the power uprate impact on HELB systems. MUR power uprate operating temperatures, pressures, and mass flow rates were compared to the analyzed conditions. The review concluded that overall, the total pipe stresses were not significantly impacted. Therefore, the MUR power uprate does not result in any new or revised pipe break locations, and the existing design basis for pipe break, jet impingement and pipe whip remains valid.

IV.1.B.vii.2 Leak Before Break Evaluation

The existing leak before break analyses justified eliminating large primary loop pipe rupture from the SPS structural design basis (Reference IV-1). The applicable pipe loadings, normal operating pressure, and temperature parameters at power uprate conditions were used to evaluate leak before break. The leak before break acceptance criteria are based on NRC Standard Review Plan, Section 3.6.3. These criteria are satisfied for primary loop piping at power uprate conditions. The recommended margins are satisfied, and the existing analyses conclusions remain valid. Therefore, the dynamic effects of RCS primary loop piping breaks are not considered in the SPS structural design basis at MUR power uprate conditions.

IV.1.B.viii LOCA Forces Including Jet Impingement and Thrust

A LOCA hydraulic forces analysis generates the hydraulic forcing functions and hydraulic loads that occur on RCS components due to a postulated LOCA. No changes in RCS design or operating pressure were made as part of the power uprate. LOCA hydraulic forces increase with lower temperatures, so they are predominantly influenced by T_{cold}. The vessel/internals, loop, and SG analyses showed acceptable results at the uprated power conditions.

IV.1.B.ix Seismic Qualification

SPS safety-related structures, systems and components are designed for both seismic and dynamic events as described in SPS UFSAR Chapter 15. The MUR power uprate impact on mechanical and electrical equipment seismic qualification, and the dynamic effects associated with pipe whip and jet impingement forces was evaluated. The mechanical and electrical equipment reviewed included equipment associated with systems essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, and preventing the significant release of radioactive material to the environment.

The primary input motions due to the design basis earthquake are not affected by the MUR power uprate. Seismic design is not impacted, because seismic requirements remain unchanged. Therefore, the seismic qualification of essential equipment supports is unaffected.

The mechanical and electrical equipment seismic qualification review demonstrated that the equipment will continue to meet the current SPS licensing basis with respect to the requirements of General Design Criteria-4; 10 CFR 50, Appendix B; and 10 CFR 100, Appendix A.

IV.1.C.i Pressurized Thermal Shock

10 CFR 50.61 Pressurized Thermal Shock (PTS) (Reference IV-10) screening calculations were performed for all Surry Unit 1 and Unit 2 reactor vessel beltline materials using neutron fluence values corresponding to the end of the current 60-year operating license (EOL). The results of these calculations were presented to the NRC in Attachment 1 of Reference IV-5. After consideration of the EOL fluence values, it was concluded that all Surry Unit 1 and Unit 2 reactor vessel beltline materials would continue to meet the 10 CFR 50.61 PTS screening criteria (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds) throughout the 60-year operating license period. Because the reactor vessel beltline neutron fluence values used in the development of the Reference IV-5 submittal conservatively bound the fluence values determined for the MUR uprated core power level, it is concluded herein that Surry Units 1 and 2 will continue to meet 10 CFR 50.61 PTS screening criteria throughout the 60-year operating license period under MUR uprated core power level conditions. Details of the evaluations that support this conclusion are presented below. In addition, Section IV.1.C.ii presents information regarding the conservatism of the reactor vessel beltline neutron fluence values used in the reactor vessel integrity evaluations at MUR uprated core power conditions, as well as information regarding the conformance of fluence analyses to the requirements of RG 1.190 (Reference IV-4).

The reactor vessel integrity analysis results presented to the NRC in Reference IV-5 were analyzed using the revised initial (unirradiated) RT_{NDT} and initial RT_{NDT} uncertainty (σ_I) values documented in the NRC-approved Topical Report BAW-2308 Revision 1-A (Reference IV-11). Since the issuance of BAW-2308 Revision 1-A (Reference IV-11), the NRC has approved a revised Topical Report BAW-2308 Revision 2-A (Reference IV-12), which resulted in small changes to initial RT_{NDT} and initial RT_{NDT} uncertainty values relative to those allowed by BAW-2308 Revision 1-A. Table 9 of BAW-2308 Revision 2-A presents a comparison of the BAW-2308 Revision 1-A (Reference IV-11) and BAW-2308 Revision 2-A (Reference IV-12) initial (unirradiated) RT_{NDT} and initial RT_{NDT} uncertainty (σ_I) values. The values from Table 9 for weld wire heats used in the fabrication of the Surry Units 1 and 2 reactor vessels are reproduced in Table IV-1.

Table IV-1
Weld Wire Heat Comparison

Linde 80 Heat	BAW-2308 Rev. 1-A Values		BAW-2308 Rev. 2-A Values		Increase (or Decrease) in Initial RT_{NDT} (°F)	Increase (or Decrease) in σ_I (°F)
	Initial RT_{NDT} (°F)	σ_I (°F)	Initial RT_{NDT} (°F)	σ_I (°F)		
299L44	-81.8	11.6	-74.3	12.8	7.5	1.2
72445	-72.5	12.3	-72.5	12.0	0	-0.3
Other heats	-47.6	17.2	-48.6	18.0	-1.0	0.8

The implications of the changes identified in Table IV-1 for Surry Units 1 and 2 are as follows:

1. The initial RT_{NDT} for Linde 80 weld heat 299L44 increased by 7.5°F (from -81.8°F to -74.3°F), and the initial RT_{NDT} uncertainty (σ_I) increased by 1.2°F (from 11.6°F to 12.8°F). The Surry Unit 1 Lower Shell Longitudinal Weld L2, weld ID SA-1526, is fabricated with weld wire heat number 299L44. The conservatism of reactor vessel integrity assessments for SA-1526 that utilized initial RT_{NDT} and uncertainty values from BAW-2308 Revision 1-A (Reference IV-11) will be explicitly examined herein in consideration of the implications of BAW-2308 Revision 2-A (Reference IV-12).

2. The initial RT_{NDT} value for Linde 80 weld heat 72445 did not change, and the initial RT_{NDT} uncertainty (σ_I) decreased by 0.3°F (from 12.3°F to 12.0°F). The Surry Unit 1 Intermediate to Lower Shell Circumferential Weld, weld IDs SA-1585 (ID 40%) and SA-1650 (OD 60%); and the Surry Unit 2 Intermediate Shell Longitudinal Weld L3 (100%), and the

Surry Unit 2 Intermediate Shell Longitudinal Weld L4 (OD 50%) are fabricated with weld wire heat number 72445. Because the initial RT_{NDT} value for weld heat 72445 did not change, and the initial RT_{NDT} uncertainty (σ_I) decreased, the results of reactor vessel integrity assessments for SA-1585 and SA-1650 weld materials that utilized initial RT_{NDT} and uncertainty values from BAW-2308 Revision 1-A (Reference IV-11) are conservative with respect to those that would be obtained using BAW-2308 Revision 2-A (Reference IV-12), and do not require further evaluation herein.

3. The initial RT_{NDT} for "Other Heats" decreased by 1.0°F, from -47.6°F to -48.6°F, and the initial RT_{NDT} uncertainty (σ_I) increased by 0.8°F, from 17.2°F to 18.0°F. The Surry Unit 1 Intermediate Shell Longitudinal Welds L3 and L4, Weld ID SA-1494, are fabricated with weld wire heat number 8T1554. The Surry Unit 2 Intermediate Shell Longitudinal Weld L4 (ID 50%), and the Surry Unit 2 Lower Shell Longitudinal Welds L2 (ID 63%) and L1 (100%), all Weld ID WF-4, as well as the Surry Unit 2 Lower Shell Longitudinal Weld L2 (OD 37%), Weld ID WF-8, are fabricated with weld wire heat number 8T1762. Welds fabricated with weld wire heat number 8T1554 and 8T1762 all fall under the heading of "Other Heats" in Table 9 of BAW-2308 Revision 2-A. The Surry Units 1 and 2 reactor vessel materials that fall under the heading of "Other Heats" were determined in the analyses that support the Reference IV-5 submittal to be non-limiting materials in reactor vessel integrity assessments for Surry Units 1 and 2. The conclusion that these materials are non-limiting remains valid after consideration of the changes in initial RT_{NDT} and uncertainty (σ_I) identified above.

Based on this assessment of the effects of BAW-2308 Revision 2-A (Reference IV-12) on previously submitted reactor vessel integrity analyses (Reference IV-5) performed using BAW-2308 Revision 1-A (Reference IV-11), only the assessments for SA-1526/299L44 must be re-examined herein to confirm that the results meet applicable regulatory criteria, and that the identified limiting material remains limiting.

For Surry Unit 1, the limiting materials in terms of absolute value of RT_{PTS} were determined in Reference IV-5 to be the Intermediate to Lower Shell Circumferential Welds SA-1585/72445 and SA-1650/72445. For these materials, the value of RT_{PTS} is 226.5°F versus the PTS screening criterion of 300°F for circumferential welds. This calculation was performed using the Initial RT_{NDT} and uncertainty (σ_I) values from BAW-2308 Revision 1-A (Reference IV-11). Because the Initial RT_{NDT} value for weld heat 72445 is unchanged by the issuance of BAW-2308 Revision 2-A (Reference IV-12), and because the initial RT_{NDT} uncertainty (σ_I) decreased, this result remains valid and conservative after consideration of the effects of BAW-2308 Revision 2-A.

For Surry Unit 1, the limiting material in terms of margin to the applicable PTS screening criterion is the Lower Shell Longitudinal Weld SA-1526/299L44. For this material, the value of RT_{PTS} is 201.8°F versus the PTS screening criterion of 270°F for plates, forgings, and axial welds. This calculation was performed using the Initial RT_{NDT} and uncertainty (σ_I) values from BAW-2308 Revision 1-A (Reference IV-11). After consideration of the effects of BAW-2308 Revision 2 (Reference IV-12), this PTS screening calculation result increases by approximately 8.5°F. This change is insufficient to cause the PTS screening calculation result to exceed the screening criterion for longitudinal welds (i.e., 270°F), or to cause the result for SA-1526/299L44 to become more limiting in terms of absolute value of RT_{PTS} than the result for SA-1585/SA-1650/72445.

For Surry Unit 2, the limiting material in terms of the absolute value of RT_{PTS} and margin to the applicable PTS screening criterion is the Intermediate to Lower Shell Circumferential Weld R3008/0227. For this material, the value of RT_{PTS} is 236.4°F versus the PTS screening criterion of 300°F for circumferential welds. There is no revised initial RT_{NDT} or uncertainty (σ_I) value for this material provided in BAW-2308 Revision 2-A (Reference IV-12), as this is a non-Linde-80 (Rotterdam) weld material. Therefore, this result is unaffected by the issuance of BAW-2308 Revision 2-A (Reference IV-12).

In summary, the fluence values used in the analyses that support the Reference IV-5 Reactor Vessel Integrity Database have been determined to conservatively bound the more recently developed fluence analysis results that explicitly consider the effects of an MUR uprated core power level. After consideration of the effects of the NRC-approved Topical Report BAW-2308 Revision 2-A on previously submitted 10 CFR 50.61 PTS screening calculations (Reference IV-5), it has been determined that all Surry Units 1 and 2 reactor vessel beltline materials meet the 10 CFR 50.61 PTS screening criteria for operation through the end of the 60-year license period at an MUR uprated core power level.

It should be noted that the current Surry Units 1 and 2 Technical Specification RCS heatup and cooldown (pressure/temperature; P/T) limit curves, LTOPS setpoints, and LTOPS enabling temperature (T_{enable}) values are applicable to cumulative core burnups of 28.8 EFPY and 29.4 EFPY (which are reached in approximately June 2011 and March 2012) for Surry Units 1 and 2, respectively. Therefore, revised RPV integrity analyses and an RVID update, including revised 10 CFR 50.61 PTS screening calculations, that explicitly include consideration of BAW-2308 Revision 2-A (Reference IV-12) are being prepared and will be submitted to NRC in accordance with applicable regulatory requirements, including 10 CFR 50.61 and 10 CFR 50 Appendix G. This submittal is not required to support the proposed MUR core power uprate.

IV.1.C.ii Fluence Evaluation

Westinghouse performed analyses to determine the RPV neutron flux and integral neutron fluence for the Surry Power Station (SPS) Units 1 and 2 Measurement Uncertainty Recapture (MUR) uprate project. The neutron flux and integral fluence values calculated by Westinghouse under MUR conditions are demonstrated below to be bounded by (i.e., are less limiting than) those used in the current reactor vessel integrity analysis of record (AOR) for SPS provided to the NRC in Reference IV-5. Therefore, MUR evaluations documented herein utilize the more conservative fluence values of the current AOR to demonstrate compliance with RPV integrity regulatory requirements under MUR uprated core power conditions.

The RVID update provided in Reference IV-5 is based on peak fast neutron fluence (E > 1.0 MeV) values for the Surry Units 1 and Unit 2 reactor pressure vessels at the end of the 60-year license period. The AOR peak reactor vessel inner surface fluence (E> 1.0 MeV) values and the Westinghouse (W) fluence values are presented in Table IV-2 for comparison.

Table IV-2
Comparison of AOR and MUR Fluence Results

Unit	Max Fluence	Max Flux	Years Exposed
Surry Unit 1 (AOR)	5.66 E19 n/cm^2	3.45 E10 n/cm^2	48.0 EFPY
Surry Unit 2 (AOR)	5.38 E19 n/cm^2	3.45 E10 n/cm^2	48.0 EFPY
Surry Unit 1 (W)	4.50 E19 n/cm^2	3.05 E10 n/cm^2	48.0 EFPY
Surry Unit 2 (W)	4.51 E19 n/cm^2	3.05 E10 n/cm^2	48.0 EFPY

As can be seen in Table IV-2, the maximum fast neutron fluence and flux values utilized in the Reference IV-5 submittal for Surry Units 1 and 2 conservatively bound (i.e., are higher in value than) those calculated by Westinghouse for post-MUR operation (i.e., operation at 2597 MWth core power starting with Surry 1 Cycle 23 and Surry 2 Cycle 23 through the end of the 60-year license period). Therefore, the fast neutron fluence values used in the analyses that support the Reference IV-5 submittal are conservative for use in assessing the effects of operation at the MUR uprated core power level. Moreover, the reactor vessel integrity analyses documented in Reference IV-5 may be used as the basis for demonstrating compliance with applicable reactor vessel integrity regulatory requirements under MUR uprated core power level conditions.

Reference IV-5 affirms the conformance of the current AOR fluence analyses with the requirements of Regulatory Guide (RG) 1.190 (Reference IV-4). Regarding the Westinghouse fluence analyses, four reactor vessel materials surveillance capsules have been withdrawn from Surry Unit 1, and five have been withdrawn from Unit 2. Measured sensor data for three irradiated dosimetry sets per unit were reported in vendor analyses. As a validation of the Westinghouse analysis of the Surry Units 1 and 2 reactor vessel neutron exposure, the measured reaction rates were used in conjunction with the current calculated neutron spectra for each of the three withdrawn surveillance capsules as input to the NRC-approved least squares dosimetry evaluation methodology (Reference IV-3). From the comparisons drawn from Westinghouse analyses of neutron fluence (E> 1.0 MeV), the adjusted-to-calculated ratios (A/C) span a range from 1.01 to 1.15 (0.84 to 1.01) with an average A/C of 1.07± 6.9% (1σ) (0.95 ± 6.9% (1σ)) for the capsule data set for Units 1 and 2, respectively. These comparisons fall well within the ± 20% criterion specified in RG 1.190, thus supporting the validation of the current calculations for applicability to the Surry Units 1 and 2 reactor pressure vessels.

IV.1.C.iii Heatup and Cooldown Pressure/Temperature Limit Curves

RT_{NDT} calculations were performed for all Surry Unit 1 and Unit 2 reactor vessel beltline materials using neutron fluence values corresponding to the end of the current 60-year operating license (EOL). The results of these calculations were presented to the NRC in Attachment 1 of Reference IV-5. After consideration of the EOL fluence values, it was concluded that the RT_{NDT} values all Surry Units 1 and 2 reactor vessel beltline materials remain less limiting than the RT_{NDT} values used in development of Technical Specification reactor coolant system (RCS) heatup and cooldown pressure/temperature (P/T) limit curves throughout the current 60-year operating license period. As described in Section IV.1.c.ii, the reactor vessel beltline neutron fluence values used in the development of the Reference IV-5 submittal conservatively bound the fluence values determined for the MUR uprated core power level. Therefore, it is concluded herein that Surry Units 1 and 2 Technical Specification heatup and cooldown P/T limit curves remain valid and conservative for operation throughout the 60-year operating license period under MUR uprated core power level conditions. Details of evaluation that support this conclusion are presented below. In addition, Section IV.1.c.ii of this evaluation presents information regarding the conservatism of the reactor vessel beltline neutron fluence values used reactor vessel integrity evaluations at MUR uprated core power conditions, as well as information regarding the conformance of fluence analyses to the requirements of RG 1.190 (Reference IV-4).

The current Surry Units 1 and 2 Technical Specification RCS heatup and cooldown P/T limit curves, and the associated LTOPS setpoints and enabling temperature values, were developed based on ¼-T and ¾-T RT_{NDT} values of 228.4°F and 189.5°F, respectively. Reference IV-5 presents the results of RT_{NDT} calculations performed in accordance with Regulatory Guide 1.99 Revision 2 (Reference IV-6), and the guidance provided in the meeting minutes from the November 12, 1997 NRC/Industry meeting on reactor vessel integrity

(Reference IV-13), for all Surry Units 1 and 2 reactor vessel beltline materials at end of the 60-year renewed license operating period at neutron fluence values corresponding to 48.0 EFPY for Units 1 and 2. The most limiting ¼-T and ¾-T RT_{NDT} values for Surry were determined to be 222.5°F and 188.6°F, respectively (for the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld material R3008/0227). The calculations that supported the determination of these limiting RT_{NDT} values and the limiting material were performed using the Initial RT_{NDT} and uncertainty (σ_I) values from BAW-2308 Revision 1-A (Reference IV-11). The Initial RT_{NDT} and initial RT_{NDT} uncertainty (σ_I) values for weld heat R3008/0227 are unaffected by the issuance of BAW-2308 Revision 2-A (Reference IV-12), since weld heat R3008/0227 is a non-Linde-80 (Rotterdam) weld material. Further, Reference IV-5 concluded that the limiting values of ¼-T and ¾-T RT_{NDT} for SA-1526/299L44 (Surry Unit 1 Lower Shell Longitudinal Weld L2) were 171.3°F and 113.3°F, respectively. (Again, this calculation was performed using the Initial RT_{NDT} and uncertainty (σ_I) values from BAW-2308 Revision 1-A (Reference IV-11).) After consideration of the effects of BAW-2308 Revision 2 (Reference IV-12), the Reference IV-5 ¼-T and ¾-T RT_{NDT} results increase by approximately 8.5°F. This change is insufficient to cause the ¼-T and ¾-T RT_{NDT} values for SA-1526/299L44 to become more limiting than the ¼-T and ¾-T RT_{NDT} values calculated in Reference IV-5 for the Surry Unit 2 Intermediate to Lower Shell Circumferential Weld material R3008/0227. By extension, the effect of BAW-2308 Revision 2 (Reference IV-12) on the ¼-T and ¾-T RT_{NDT} values for SA-1526/299L44 is clearly insufficient to cause the limiting values of ¼-T and ¾-T RT_{NDT} for SA-1526/299L44 to exceed the values of ¼-T and ¾-T RT_{NDT} used in the development of Surry Units 1 and 2 Technical Specification RCS heatup and cooldown P/T limit curves.

In summary, the fluence values used in the analyses that support the Reference IV-5 Reactor Vessel Integrity Database have been determined to conservatively bound the more recently developed fluence analysis results that explicitly consider the effects of an MUR uprated core power level. After consideration of the effects of the NRC-approved Topical Report BAW-2308 Revision 2-A on previously submitted RT_{NDT} calculations (Reference IV-5), it has been determined that the Surry Units 1 and 2 Technical Specification heatup and cooldown P/T limit curves, and the associated LTOPS setpoints and LTOPS enabling temperature (T_{enable}) values, remain valid and conservative for operation throughout the 60-year operating license period under MUR uprated core power level conditions.

As noted in Section IV.1.C.i, current Surry Units 1 and 2 Technical Specification RCS heatup and cooldown P/T limit curves, LTOPS setpoints, and LTOPS enabling temperature (T_{enable}) values are applicable to cumulative core burnups of 28.8 EFPY and 29.4 EFPY (which are reached in approximately June 2011 and March 2012) for Surry Units 1 and 2, respectively. Therefore, revised RPV integrity analyses and an RVID update that explicitly include consideration of BAW-2308 Revision 2-A (Reference IV-12) are being prepared and will be submitted to NRC in accordance with applicable regulatory requirements, including 10 CFR 50.61 and 10 CFR 50 Appendix G. This submittal is not required to support the proposed MUR core power uprate.

IV.1.C.iv Low Temperature Overpressure Protection

The evaluation presented in Section IV.1.C.iii for heatup and cooldown limit curves also applies to LTOPS setpoints. Specifically, Section IV.1.C.iii concludes that, after consideration of the effects of the NRC-approved Topical Report BAW-2308 Revision 2-A on previously submitted RT_{NDT} calculations (Reference IV-5), the Surry Units 1 and 2 Technical Specification heatup and cooldown P/T limit curves, and the associated LTOPS setpoints and LTOPS enabling temperature (T_{enable}) values, remain valid and conservative for operation throughout the 60-year operating license period under MUR uprated core power level conditions.

IV.1.C.v Effect on Upper Shelf Energy Calculation

The evaluation of reactor vessel fast neutron fluence presented in Section IV.1.C.ii above demonstrates that fluence values used in the Reference IV-5 reactor vessel integrity assessments conservatively bound those calculated by Westinghouse, which explicitly consider the effects of the MUR uprate. Further, 10 CFR 50 Appendix G Upper Shelf Energy calculations are not dependent on the Initial RT_{NDT} and uncertainty (σ_I) values from BAW-2308 Revision 1-A (Reference IV-11) or Revision 2 (Reference IV-12). Therefore, the 10 CFR 50 Appendix G Upper Shelf Energy calculation results presented in Reference IV-5 remain valid and conservative for operation throughout the 60-year operating license period under MUR uprated core power level conditions. The results in Reference IV-5 demonstrate acceptable Upper Shelf Energy for all Surry Units 1 and 2 reactor vessel beltline materials through calculations that conform with the requirements of RG 1.99 Revision 2 (Reference IV-6), or through an equivalent margins analysis.

IV.1.C.vi Surveillance Capsule Withdrawal Schedule

NRC-approved reactor vessel materials surveillance capsule withdrawal schedules for Surry Units 1 and 2 are presented in the Updated Final Safety Analysis Report (UFSAR). This schedule calls for periodic withdrawal of the surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The surveillance capsule withdrawal schedules presented in the UFSAR were developed using American Society for Testing and Materials (ASTM) Standard E185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels* (Reference IV-7). The schedule for withdrawing reactor vessel materials surveillance capsules is dependent on the calculated fast neutron fluence to the surveillance capsules relative to the calculated fluence to the reactor vessel beltline (i.e., the capsule "lead factor"). Surveillance capsule lead factors are only weakly dependent on core power level. Therefore, the proposed MUR uprated power level has a negligible effect on the schedule for withdrawal of surveillance capsules.

Revised reactor vessel materials surveillance capsule withdrawal schedules to accommodate the 60-year license period have been developed and submitted to the NRC under separate cover for review and approval in accordance with 10 CFR 50 Appendix H, Section III.B.3. The proposed schedules satisfy the requirements and guidance of ASTM E-185-82 (Reference IV-7) and the Generic Aging Lessons Learned (GALL) Report (NUREG-1801) (Reference IV-8) for surveillance capsule withdrawal and testing. With that submittal, Surry Units 1 and 2 will remain in compliance with the requirements of 10 CFR 50 Appendix H for operation during the 60-year extended license period. That submittal is not required to support the proposed MUR core power uprate.

IV.1.D Codes of Record

Table IV-3
Codes of Record

Component	Code	Code Class	Edition and Addenda
Reactor Vessel(1,2)	ASME III	A	1968 Edition through Winter 1968 Addenda
CRDM	ASME III	A	Unit 1 - 1965 Edition through Summer 1966 Addenda Unit 2 - 1995 Edition through 1996 Addenda
Steam Generator			
Tube side	ASME III	A	1974 Edition through Winter 1976 Addenda
Shell side(3)	ASME III	C	1974 Edition through Winter 1976 Addenda
Pressurizer	ASME III	A	1965 Edition through Winter 1965 Addenda
Reactor Coolant System			
Valve, fittings and piping	USAS B31.1(4)	1	1955 Edition
Loop Stop Valves	ASME III		1968 Edition through Summer 1968 Addenda
Safety valves	ASME III	A	1977 Edition through Summer 1977 Addenda
Reactor coolant pump	No code (design per ASME III - Article 4)		
BOP Piping	ANSI B31.1.0		1967 Edition

1. Unit 1 reactor vessel closure head was replaced with closure head designed, fabricated and manufactured to French Construction Code (R-CCM) 1993 Edition with, 1st Addenda June 1994, 2nd Addenda June 1995, 3rd Addenda June 1996 and modification sheets FM 797, 798, 801, 803 through 807. The sizing calculations and the stress and fatigue analysis were performed to ASME B&PV Code, Section III, 1995 Edition 1996 Addenda. The Design Report certified that the Unit 1 closure head meets the design requirements and stress limits for the ASME B&PV Code, Section III, 1968 Edition through Winter 1968 Addenda.
2. Unit 2 reactor vessel closure head was replaced with a closure head fabricated and manufactured by Mitsubishi Heavy Industries. The replacement reactor vessel closure head was designed and fabricated in accordance with ASME B&PV Code, Section III, 1995 Edition with 1996 Addenda. The stress and fatigue analyses were performed to ASME B&PV Code, Section III, 1995 Edition with 1996 Addenda.
3. The shell side of the SG conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.
4. A reanalysis of the pressurizer surge line to account for the effect of thermal stratification and striping was performed in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III 1986 with addenda through 1987 incorporating high cycle fatigue as required by NRC Bulletin 88-11, dated December 20, 1988.

There are no changes to the codes of record listed above in Table IV-3.

IV.1.E Changes to Component Inspection and Testing Programs

IV.1.E.i Inservice Testing Program

10 CFR 50.55a(f), *Inservice Testing Requirements*, mandates the development and implementation of an IST Program. SPS has developed and is implementing an IST Program for pumps and valves per the applicable requirements. SPS Technical Specification 6.4.I describes the surveillance requirements that apply to the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

The applicable system analyses were reviewed to determine if the MUR power uprate would impact the existing IST Program. There are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect component performance or test acceptable criteria. Therefore, the MUR power uprate has no impact on the testing required by the IST Program.

IV.1.E.ii Inservice Inspection Program

10 CFR 50.55a(g), *Inservice Inspection Requirements,* mandates the development and implementation of an ISI Program. The applicable program requirements are specified in ASME B&PV Code, Section XI. SPS has developed and is implementing an ISI Program per these requirements. UFSAR Section 4.4.1.7 describes the ISI Program.

This evaluation reviewed the MUR power uprate impact on the existing ISI Program. System classifications or boundaries for ASME Class 1, 2, and 3 systems are not affected. Inspection frequencies and required procedures for ASME Class 1, 2, and 3 components and their supports as described in the ISI Program Manual are not affected. Therefore, the MUR power uprate has no impact on the existing ISI Program.

IV.1.E.iii Erosion/Corrosion Program

SPS has established and maintains a FAC Program per NRC Generic Letter 89-09, *Erosion/Corrosion – Induced Pipe Wall Thinning.* The FAC Program meets the intent of EPRI NSAC-202L, *Recommendations for an Effective Flow-Accelerated Corrosion Program,* and INPO EPG-06, *INPO Engineering Guide – Flow Accelerated Corrosion.* This program provides a standardized method of identifying, inspecting, and tracking components susceptible to FAC wear in both single and two-phase flow conditions. Program elements include: FAC susceptibility analysis and modeling, FAC inspection and evaluation, operational experience reviews, and crossover/crossunder main steam piping and moisture separators/reheaters inspections and evaluations. In general, plant systems are considered susceptible to FAC unless excluded by defined criteria.

The criteria includes: material, moisture content, temperature, dissolved oxygen, frequency of system usage, plant-specific operating experience, and industry operating experience. SPS utilizes the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code to assist in predicting and tracking FAC susceptible components. The CHECWORKS SFA computer code has been used to create unit-specific databases. Once the database has been built, the application is used to perform analysis and data interpretation. These analytical models result in Wear Rate Analysis that rank components in order of predicted FAC wear and predicted time to reach minimum code wall thickness. In order to evaluate the power uprate impact on FAC wear rates, the SPS Unit 1 and 2 CHECWORKS SFA models were updated to incorporate the changes associated with the power uprate.

SPS Unit 1 and 2 evaluations were performed to determine the impact on remaining service life as a result of the increase in wear rates due to the MUR power uprate. Tables IV-4 and IV-5 summarize these reviews.

Table IV-4
Surry Unit 1 Wear Rate Analysis

Model	System	Increase in Wear Rate	Decrease in Time to T_{crit} (code wall)	Notes
4th Point Extraction Pipe	Extraction Steam	36.6%	27.9%	Exceeds remaining plant life.
4th Point Extraction Elbow	Extraction Steam	29.9%	26.2%	Exceeds remaining plant life.
5th Point Extraction Elbow	Extraction Steam	16.7%	26.3%	Subject to appropriate inspection.
6th Point Extraction Pipe	Extraction Steam	4.0%	5.2%	Exceeds remaining plant life.
6th Point Extraction Pipe	Extraction Steam	5.1%	5.0%	Exceeds remaining plant life.

Table IV-5
Surry Unit 2 Wear Rate Analysis

Model	System	Increase in Wear Rate	Decrease in Time to T_{crit} (code wall)	Notes
4th Point Extraction Reducer	Extraction Steam	12.3%	11.8%	Exceeds remaining plant life.
4th Point Extraction Elbow	Extraction Steam	13.1%	12.8%	Exceeds remaining plant life.
5th Point Extraction Elbow	Extraction Steam	16.0%	17.5%	Subject to appropriate inspection.
5th Point Extraction Pipe	Extraction Steam	16.9%	11.0%	Subject to appropriate inspection.
HP Feedwater Drain Elbow	Steam Drain	11.4%	12.3%	Exceeds remaining plant life.

Tables IV-4 and IV-5 represent randomly selected piping components in the five systems expected to experience the greatest increase in FAC wear as a result of the MUR power uprate. The randomly selected piping components in other systems have a smaller increase or an actual decrease in FAC wear.

Upon power uprate implementation, the CHECWORKS SFA databases for SPS Units 1 and 2 will be updated and validated. The wear rate analysis models will be analyzed using the updated information and the Wear Rate Analysis – Service Life Report for each model will be reviewed. Any piping components with a low or a negative time for remaining service life will be evaluated for a future inspection.

Based on the reviews conducted for the impact of increased wear rates on remaining service life, there is no significant impact. No additional secondary system lines were identified as requiring monitoring for FAC wear as a result of the MUR power uprate. The remaining service life for the modeled FAC susceptible lines will continue to be monitored and will be documented at the end of each refueling outage.

IV.1.F Impact of NRC Bulletin 88-02, *Rapidly Propagating Fatigue Cracks in Steam Generator Tubes*

NRC Bulletin 88-02 required actions by operating license holders of Westinghouse designed nuclear power reactors with SGs having carbon steel support plates. SGs in this category include Westinghouse models 13, 27, 44, 51, D1, D2, D3, D4 and E. These actions were required to minimize the potential for a SGTR caused by rapidly propagating fatigue cracks such as occurred at North Anna 1 on July 15, 1987. The tube rupture was caused by high cycle fatigue.

As previously stated, SPS Units 1 and 2 Model 51 SGs were replaced in 1981 and 1980, respectively. The Model 51F replacement SGs are a blend of a new tube bundle, lower shell and primary channel head region, and primary moisture separators and feed rings, with the original upper shell and secondary moisture separator (Model 51 steam drum). An evaluation was performed on the potential for high cycle fatigue in unsupported SG U-bend tubes. One of the prerequisites for high cycle SG U-bend fatigue is a dented support condition at the upper plate. This support condition results from corrosion product build-up associated with drilled holes in carbon steel tube support plates. Since the broached stainless steel support plate in this model SG is designed to inhibit the introduction of corrosion products, the support condition (i.e., denting) necessary for high cycle fatigue should not occur. Dominion has not observed any corrosion product build-up to date. Therefore, high cycle fatigue associated with unsupported inner row SG tubes is not a concern in this model SG.

IV REFERENCES

IV-1 WCAP-15550, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for the Surry Units 1 and 2 Nuclear Power Plants for the License Renewal Program,* August 2000.

IV-2 WCAP-14040-NP-A, Revision 4, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,* May 2004.

IV-3 WCAP-16083-NP-A, Revision 0, *Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,* May 2006.

IV-4 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,* U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March 2001.

IV-5 Letter from E.S. Grecheck (Dominion) to USNRC, *Surry Power Station Units 1 and 2 Update to Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Basis per 10 CFR 50.60(b),* Serial Number 06-434, June 13, 2006.

IV-6 Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials,* May 1988.

IV-7 American Society for Testing and Materials (ASTM) E185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.*

IV-8 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* July 2001.

IV-9 EPRI Report MRP-146, *Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines,* 1011955, June 2005.

IV-10 Title 10, Code of Federal Regulations, Part 50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.*

IV-11 Areva Report, BAW-2308, Revision 1-A, *Initial RT_{NDT} of Linde 80 Weld Materials,* August 2005.

IV-12 Areva Report BAW-2308, Revision 2-A, *Initial RT_{NDT} of Linde 80 Weld Materials,* March 2008.

IV-13 Memorandum from K. R. Wichman to E. J. Sullivan, *Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92-01, Revision 1, Supplement 1 Responses,* November 19, 1997.

V. ELECTRICAL EQUIPMENT DESIGN

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

   A. emergency diesel generators

   B. SBO equipment

   C. EQ of electrical equipment

   D. grid stability

RESPONSE TO V - ELECTRICAL EQUIPMENT DESIGN

V.1.A Emergency Diesel Generators

The EDG system provides a safety-related emergency source of AC power for the engineered safeguards and selected BOP emergency loads, in the event that the normal AC power is interrupted. The EDG system consists of three 100 percent capacity EDGs for the two units. One EDG is dedicated to Unit 1 and supplies emergency power to the 1H emergency bus. The second EDG is dedicated to Unit 2 and supplies emergency power to the 2H emergency bus. The third EDG functions as a backup to either Unit 1 or Unit 2 and feeds either the 1J or 2J emergency bus.

The electrical loads that changed as a result of the power uprate are not fed from the EDG system. There are no increases to the emergency buses loads supported by the EDGs. The EDG system equipment capacity and capability for plant operation at the uprate conditions are bounded by the EDG loading tables. The EDG loading tables are supported by the existing analysis of record. Both the bounding analysis and the EDG loading tables demonstrate that the EDG system has adequate capacity and capability to provide onsite standby power for safety-related loads following a LOOP with or without a concurrent accident. Therefore, the EDG system is not affected by the MUR power uprate.

V.1.B Station Blackout Program

10 CFR 50.63 requires each light water cooled nuclear power plant to withstand and recover from a loss of all AC power, referred to as SBO. The SPS coping duration is four hours. This is based on an evaluation of the offsite power design

characteristics, emergency AC power system configuration, and EDG reliability. The evaluation was completed per NUMARC 87-00 and NRC Regulatory Guide 1.155. The MUR power uprate has no impact on the current SBO coping duration of four hours. The MUR power uprate was evaluated for impact on the alternate AC power source and the following SBO coping issues: emergency condensate storage tank inventory, Class 1E battery capacity, ventilation, compressed air, and containment isolation.

V.1.B.i Alternate AC Power Source

The alternate AC power source consists of a diesel generator and support subsystems (e.g., starting air, cooling water, lubrication and fuel oil). The alternate AC diesel generator, with its separate fuel supply, can be aligned to the Unit 1 J Bus or the Unit 2 H Bus. This provides additional assurance that AC power will remain available. The alternate AC diesel generator has sufficient capacity to operate systems necessary for coping with a SBO event for the required coping period.

V.1.B.ii Emergency Condensate Storage Tank Inventory

The ECST provides adequate inventory to maintain a SPS unit in hot shutdown for eight hours at MUR power uprate conditions. Since SPS has a four-hour SBO coping period, the ECST provides adequate inventory for decay heat removal following a SBO event at uprated conditions. The SBO analysis assumes 2597 MWt, which is 102% of 2546 MWt.

V.1.B.iii Class 1E Battery Capacity

SBO is a four-hour event at Surry, and it is assumed that no EDGs are available. The power to two emergency buses (1J and 2H) is restored within 10 minutes via the alternate AC power source, which restores power to the chargers for Class 1E station batteries 1B and 2A. The 10-minute discharge scenario for these two batteries is bounded by the two-hour design basis accident duty cycle. Power to the chargers for the other two station batteries (1A and 2B) must be restored within four hours. Station Batteries 1A and 2B have been evaluated for this four-hour SBO discharge scenario and were determined to have adequate capacity. The MUR power uprate does not affect any DC powered indication, control, or protection equipment. Therefore, the Class 1E batteries are acceptable at MUR power uprate conditions.

V.1.B.iv Ventilation

Evaluations have been performed for the following areas containing SBO equipment: AFW pump house, charging pump cubicles, control room, emergency switchgear rooms, and containment. The turbine driven AFW pump room, charging pump cubicles, control room, and emergency switchgear rooms are unaffected by the MUR power uprate. The containment pressure and temperature resulting from a LOCA or MSLB envelope the SBO event at MUR power uprate conditions.

V.1.B.v Compressed Air

The power uprate does not affect the capability for manual operation of air-operated valves, or the capability to restore compressed air by powering an instrument air compressor immediately from the unaffected unit or within one hour on the SBO unit.

V.1.B.vi Containment Isolation

The power uprate does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is not related to power level. The evaluation for containment isolation at current plant conditions remains applicable at MUR power uprate conditions

V.1.C Environmental Qualification of Electrical Equipment

The term EQ applies to equipment important-to-safety. The intent is to ensure this equipment remains functional during and following design basis events. The SPS EQ Program has been developed to ensure that EQ criteria are applied to electrical equipment important-to-safety as specified in 10 CFR 50.49, and to document the process used to demonstrate this qualification. Surry Units 1 and 2 are licensed to implement the 10 CFR 50.49 requirements through DOR Guidelines and IEEE 323-1974 (Reference V-2).

There is no effect on EQ related non-radiological conditions (e.g., temperature, pressure, humidity) resulting from the MUR power uprate. The source terms used for the radiation aspects of the EQ program evaluations have been adjusted as described in Section III.i to accommodate MUR power uprate operation.

Radiation dose qualification is based on the sum of the normal operational dose plus the accident dose. The increase in the post-accident integrated dose conservatively determined for the power uprate (Section III.1) has been evaluated for all equipment in affected environmental zones. The evaluation in Section III.1 (summarized in Table III-1) indicates that the increased radiation levels in some zones may impact equipment qualification for certain classes of equipment. Disposition of these specific cases is presented below.

<u>Ex-core Neutron Detectors</u>

The ex-core neutron detectors are scheduled to be replaced during the fall 2010 Unit 1 outage and the spring 2011 Unit 2 outage. The equipment replacement is a result of existing EQ Periodic Maintenance schedules. Evaluation of the radiation analysis has not been developed. Prior to operating above the current RP of

2546 MWt, Dominion will incorporate changes in the qualified lifetime of this equipment into EQ program documentation.

Hydrogen Recombiners and Hydrogen Monitoring Equipment

An NRC Safety Evaluation was included in Amendment Nos. 239 and 238 to Renewed Facility Operating License Nos. DPR-32 and DPR-37for the Surry Power Station, Unit Nos. 1 and 2, dated March 22, 2005 (ADAMS Accession No. ML050840168). This Safety Evaluation determined that the Hydrogen Monitoring Equipment no longer meets the definition of a safety-related component as defined in 10 CFR 50.2. As a result, hydrogen monitors will be removed from the Surry EQ program and do not need to be evaluated.

Dominion has reviewed the effects of the proposed power uprate on the EQ of electrical equipment, and concludes that the evaluation has adequately addressed the effects of the proposed power uprate on the environmental conditions for the qualification of electrical equipment. Based on this evaluation, the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed power uprate. Therefore, Dominion finds the proposed MUR power uprate acceptable with respect to the EQ of electrical equipment.

V.1.D Grid Stability

V.1.D.i Background

SPS currently has a MVAR output limitation due to the 4 kV station service buses. The station service buses have a maximum voltage of 4.4 kV. The 941.7 MVA main generators at Surry have been replaced with 1055 MVA generators and associated exciters and voltage regulators. The generators are capable of producing approximately 480 MVARs. However, because station service bus has a maximum voltage of 4.4 kV, the generator output is limited to 400 MVARs out or 200 MVARs in. Dominion assessed the impact of a 180 MWe (i.e., 90 MWe per unit), of new generation capacity on the Dominion transmission system. Dominion is anticipating additional plant modifications that would result in additional electrical power increases beyond that proposed by this MUR LAR. Grid stability studies were conducted assuming that power increases were in effect, so the results bound the MUR power uprate. The transmission system assessment was based on Pennsylvania, New Jersey, Maryland Interconnection's (PJM) best assumptions at the present time for load growth and new generation through the summer of 2012. The evaluation included load flow studies of import/export system conditions and single-contingency, both normal and stressed, system conditions. Short circuit duty screening was not required due to no changes in existing equipment. A stability analysis was performed. Dominion considers a transmission facility overloaded if it exceeds 94% of its emergency rating under normal and stressed conditions.

V.1.D.ii Proposed New Generation Impact Analysis

Dominion routinely evaluates the impact that a proposed new generation resource will have under maximum generation conditions and stressed system conditions. Two different assessments were conducted: local generation and import/export conditions.

The local generation study assessed station operation at maximum capability. The study identified no transmission deficiencies. The import/export study assessed conditions into and out of the Dominion system. Any new facility interconnected with the Dominion system should not significantly decrement First Contingency Incremental Transfer Capability between utilities. The study indicated no decrement to system First Contingency Incremental Transfer Capability. In the summary section of the PJM system impact studies, the current MFO is 842 MWe for each unit. The MUR power uprate will increase each unit's generating capacity by approximately 15 MWe.

The PJM system impact studies (SIS) describe the final power output values used in the stability analysis. Increased generation requests for MUR and MUR plus turbine replacement are stated for each unit. Gross and net generator output MWe values used in the PJM impact and stability studies are bounding values that will not be exceeded during actual operation. These MWe values include expected additional MWe output due to MUR power increase, plus more efficient steam turbines, with additional margin. The design inputs consist of 28 MWe for house loads, 15 MWe for MUR, and 75 MWe for turbine replacement. The gross generator output, for each unit, is derived by adding the house loads and the MUR loads to the MFO of 842 MWe for each unit. The MFO values are based on maximum winter generation output and modeled as follows:

Queue Unit # Case: Gross Generator Output = MFO + House Loads + Case Increase

S111 Unit 2 MUR only: 885 MWe = 842 MWe + 28 MWe + 15 MWe
S113 Unit 1 MUR only: 885 MWe = 842 MWe + 28 MWe + 15 MWe
S114 Unit 1 MUR + turbine replacement: 960 MWe = 842 MWe + 28 MWe + 90 MWe
S115 Unit 2 MUR + turbine replacement: 960 MWe = 842 MWe + 28 MWe + 90 MWe

The SIS concluded that no transient stability issues related to the Unit 1 and Unit 2 upgrades were found. The SIS further concluded that no transmission deficiencies were identified and no decrement to system First Contingency Incremental Transfer Capability between utilities was indicated.

As described above, PJM uses the MFO plus station loads to determine the facility gross output for stability analyses. PJM uses summer net generation capability to determine thermal system impacts. Station auxiliary loads are added to the MFO for stability analysis. PJM uses maximum winter unit output at light load conditions to generate worst-case stability conditions. The MFO is added to station auxiliary loads to determine generator gross output for stability. The gross output is used in the dynamic stability analysis, which monitors rotor angle, terminal voltage, field voltage, electrical power, and speed deviation. PJM's stability analyses monitor the key variables to ensure post-trip that these variables are maintained within acceptable limits. Stability is performed using Power System Simulator for Engineering software, which uses industry accepted mathematical modeling methods, to ensure that system voltages (and other variables) are maintained through the transient and post-transient. PJM analysis is intended to assess unit behavior given external system disturbances. For external unit trip disturbances of either or both units, house loads are maintained as loads on the grid. The PJM studies considered both conditions - MUR only and MUR plus turbine replacement. Both analyses are contained in the documents and, based on the results of the studies, interconnection service agreements (ISA) were established at 1864 MWe (MFO–House Loads), for both units of the facility. The new ISA will permit Dominion to perform the MUR uprate followed by the turbine replacement efficiency uprate.

V.1.D.iii Stability Analysis

The range of contingenies evaluated was limited to that necessary to assess compliance with the Dominion criteria. Two types of faults were considered in this study: three-phase faults with primary clearing time and stuck breaker fault followed by another single line to ground fault. No secondary protection faults were tested due to the presence of dual primary relays in area of study.

No transient stability issues related to the SPS power uprate were identified. Therefore, the current grid configuration and capacity is adequate to handle the additional megawatts generated from the MUR power uprate. The details supporting the system stability for SPS are contained in the PJM Generator Impact Study. The study contains the system impacts, power flow studies, network conditions, and supporting one-line diagrams. The system impact study and the interconnect service agreement are available on the PJM website and are identified under the Queue numbers.

V.1.E Onsite Power Systems

The AC Distribution System is the source of power for the non safety-related buses and the safety-related emergency buses. It consists of the 4.16 kv, 480 v, and 120 v systems (excluding the EDGs). The electrical changes resulting from the MUR power uprate occur in equipment primarily at the 4.16 kv voltage level. The following loads were affected by the uprate: main feedwater pumps, condensate pumps, LP heater drain pumps, HP heater drain pumps, bearing cooling pumps, and RCPs. With the exception of the RCPs, none of these revised brake horsepower values exceeded the motor nameplate rating, although the operating points changed. An evaluation was performed that determined the increased RCP brake horsepower for MUR power uprate conditions is acceptable. An evaluation also determined that current loading levels under MUR power uprate conditions are bounded by the 4.16 kv buses existing capability. There were no load increases on the 480 v buses. The 120 v system loads are not related to the power generation process and are therefore independent of the MUR power uprate. The 125 vdc system loads are also not related to the power generation process and are therefore independent of the MUR power uprate. The AC 4.16 kv, 480 v, 120 v and DC 125 v electrical distribution systems are acceptable at power uprate conditions.

V.1.F Power Conversion Systems

As a result of the MUR power uprate, the RP will increase from 2546 MWt to 2587 MWt. This increase in thermal power will result in an increase in electrical power output, which affects power block equipment.

V.1.F.i Main Generator

Unit 1

The nameplate rating is 1055 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 400 MVARs out or 200 MVARs in, and maintain generator load and hydrogen pressure within the limits of the Generator Calculated Capability Curve with a generator rating of 1055.0 MVA. The main generator output at the current NSSS power level of 2555 MWt is 850.2 MWe. The anticipated main generator output is 864.7 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 864.7 MWe, the generator is capable of exporting 500 MVAR (lagging power factor of 0.865) and importing approximately 430 MVAR (leading power factor of 0.899). However, the 864.7 MWe for Unit 1 is a gross MWe value and does not take into account the approximate 28 MWe of internal electrical loads the plant represents to the generator output for each unit. Subtracting the 28 MWe of internal electrical loads from the Unit 1 heat balance value of 864.7 MWe yields a net MFO of 836.7 MWe, which is below the 857 MWe value in the PJM study (Attachment 5, Section V.1.D.ii). The exciter has the capability to support main generator operation within its restricted operational rating and within the capability curve for leading and lagging power factors. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability and the MFO for Unit 1 is still bounded by the PJM studies.

Unit 2

The nameplate rating is 1055 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 400 MVARs out or 200 MVARs in, and maintain generator load and hydrogen pressure within the limits of the Generator Calculated Capability Curve with a generator rating of 1055.0 MVA. The main generator output at the current

NSSS power level of 2555 MWt is 850.7 MWe. The anticipated main generator output is 865.6 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 865.6 MWe, the generator is capable of exporting 500 MVAR (lagging power factor of 0.865) and importing approximately 430 MVAR (leading power factor of 0.899). However, the 865.6 MWe for Unit 2 is a gross MWe value and does not take into account the approximate 28 MWe of internal electrical loads the plant represents to the generator output for each unit. Subtracting the 28 MWe of internal electrical loads from the Unit 2 heat balance value of 865.6 MWe yields a net MFO of 837.6 MWe, which is below the 857 MWe value in the PJM study (Attachment 5, Section V.1.D.ii). The exciter has the capability to support main generator operation within its restricted operational rating and within the capability curve for leading and lagging power factors. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability and the MFO for Unit 2 is still bounded by the PJM studies.

V.1.F.ii Isolated Phase Bus

The isophase bus is rated for 26,000 amps. The MUR power increase will raise the isophase bus current to approximately 25,214 amps for Unit 1 and 25,240 amps for Unit 2. Therefore, the increase from the MUR power uprate remains below the isophase bus maximum capability.

V.1.F.iii Main (Step-up) Transformers

The main transformers increase the main generator 22 kv output voltage to the 230 kv transmission voltage for Unit 1 and 500 kv transmission voltage for Unit 2. These transformers are rated for 1200 MVA, which is above the main generator 1055 MVA output capability. The transformers are sized to handle the MUR power uprate conditions given that the uprated loadings of the main transformers are 916 MVA minus the station service transformer loadings.

V.1.F.iv Unit Station Service Transformers

The unit station service transformers are supplied by the 22 kv isolated phase bus and power the 4.16 kv switchgear, 480 v load centers, and motor control centers during normal operating conditions. The 4.16 kv normal switchgear buses are transferred and connected directly to the secondary of the reserve station service transformers during station startup and shutdown conditions. The BOP electrical loads affected by the uprate increase the loading on the unit station service transformers. The uprated loadings of the station service transformers follow:

Unit 1	Unit 2
SST 1A: 15.76 MVA	SST 2A: 15.48 MVA
SST 1B: 14.88 MVA	SST 2B: 14.16 MVA
SST 1C: 15.83 MVA	SST 2C: 15.61 MVA

Even with the increased load, the unit station service transformers remain within their current rating.

V.1.F.v Reserve Station Service Transformers

The reserve station service transformers are supplied by the 34.5 kv switchyard and 4.16 kv transfer buses. The BOP electrical loads affected by the uprate increase the loading on the reserve station service transformers. The uprated loadings of the reserve station service transformers follow:

RSST A: 20.49 MVA
RSST B: 22.40 MVA
RSST C: 25.80 MVA

Even with the increased load, the reserve station service transformers remain within their current rating.

V.1.G Switchyard

The current to the switchyard is bounded by the main transformers' capability. The overhead lines from the main transformers to the switchyard are capable of carrying the full transformer load. Therefore, the overhead lines are acceptable at the MUR conditions. An evaluation determined that the small increase in power output does not significantly impact the switchyard equipment. The switchyard system analyses bound the MUR power uprate conditions.

V REFERENCES

V-1 NRC Bulletin 79-01, *Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors.*

V-2 IEEE Standard 323-1974, *Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.*

VI. SYSTEM DESIGN

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems.

    A. NSSS interface systems for pressurized water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling water reactors (BWRs) (e.g., suppression pool cooling), as applicable

    B. containment systems

    C. safety-related cooling water systems

    D. spent fuel pool storage and cooling systems

    E. radioactive waste systems

    F. engineered safety features (ESF) heating, ventilation and air conditioning

RESPONSE TO VI - SYSTEM DESIGN

VI.1.A Interface Systems

VI.1.A.i Main Steam System

The MSS is described in UFSAR Section 10.3. This system was evaluated to determine the impact of the MUR power uprate. Component parameters are bounded by the original design equipment ratings, or by the original design considerations for off-normal operation. Therefore, the MSS is acceptable at power uprate conditions.

VI.1.A.i.a Main Steam Piping

MSS pressures, temperatures and velocities were evaluated. System pressures and temperatures are bounded by piping design parameters during power uprate conditions, with the exception of an insignificant pressure increase in the crossover piping and associated relief header piping. The velocities were bounded by the maximum recommended velocities. Main steam piping is acceptable at MUR power uprate conditions.

VI.1.A.i.b Main Steam Safety Valves

A total of five ASME B&PV Code MSSVs are located on each main steam line outside reactor containment and upstream of the MSTVs. MSSV lift setpoints are determined by SG design pressure and the ASME B&PV Code. The SG design pressure has not changed with the MUR power uprate, so the existing MSSV setpoints are unchanged. Main steam overpressure events have been analyzed and the MSSVs are adequate for the MUR power uprate.

VI.1.A.i.c Main Steam Trip Valves and Non-Return Valves

The MSTVs provide a means to isolate an SG in the event of a downstream steam line rupture. The non-return valves are located downstream of the MSTVs and prevent reverse flow in the main steam lines. The MSTVs are required to close within five seconds (stroke time) in the event of a main steam line break. The power uprate does not affect the MSTVs' ability to close within the required time period. Design loads and associated stresses resulting from rapid valve closure do not change with the power uprate. The MUR power uprate steam flow is bounded by the maximum steam flow for the non-return valves. The worst case for differential pressure increase is controlled by the steam line break areas, SG flow restrictor throat area, valve seat bore, and no load operating pressure. Since the power uprate does not impact these variables, the maximum pressure design loads and associated stresses resulting from MSTV and non-return valve rapid closure will not change. The maximum differential pressure requirement remains satisfied. Therefore, the MSTVs and non-return valves are acceptable at MUR power uprate conditions.

VI.1.A.i.d Moisture Separator Reheaters

Shell side and tube side pressures remain bounded by the MSR design conditions at power uprate conditions. The MSR safety valves are capable of passing the required load and are bounded by the valves' calculated design capacity.

VI.1.A.ii Steam Dump

The SPS steam dump function is accomplished by the SG PORVs (atmospheric relief valves) and the steam dump system (turbine bypass valves). The SG PORVs are described in UFSAR Section 10.3. The steam dump system is described in UFSAR Sections 7.3 and 10.3.

VI.1.A.ii.a Steam Generator PORVs

There are three SG PORVs per unit, one on each MS line. The SG PORVs are located upstream of the MSTV and adjacent to the MSSV. There is no change in function associated with the power uprate. The SG PORVs automatically modulate open and exhaust to the atmosphere whenever the steam line pressure exceeds a predetermined setpoint. This minimizes safety valve lifting during steam pressure transients. The SG PORV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures change for the proposed range of NSSS operating parameters, the SG PORV setpoint is unchanged.

The primary function of the SG PORVs is to provide a means for decay heat removal and plant cooldown when the condenser, the condenser CW pumps, or steam dump to the condenser is not available. The SG PORVs are sized to have a capacity equal to approximately 10% of rated steam flow at no-load pressure. The SG PORVs have a capacity of 10% at uprated conditions. Therefore, the SG PORVs are acceptable for operation at uprate conditions.

VI.1.A.ii.b Steam Dump System

The steam dump system creates an artificial steam load by dumping steam to the main condenser. Each Surry unit is provided with eight condenser steam dump valves. Steam dump in conjunction with the reactor control system permits the NSSS to withstand an external load reduction of up to 50% of plant-rated electrical load without a reactor trip. The NSSS control systems margin-to-trip analysis confirmed the steam dump system capability at uprated power conditions. There is acceptable margin to the relevant reactor trip setpoints during and following the 50% load rejection transient. To provide effective flow control on large step-load reductions or a plant trip, the steam dump valves are required to go from full-closed to full-open in three seconds at any pressure between 50 psi less than full-load pressure and SG design pressure. The steam dump valves are also required to modulate to control flow. The steam dump valves continue to satisfy these requirements at MUR power uprate conditions.

VI.1.A.iii Extraction Steam System

The extraction steam system heats the condensate and feedwater at various stages prior to the SGs, and provides the normal steam supply to the auxiliary steam system. Based on evaluation results, the extraction steam system operating parameters (pressure, temperature, flow, velocity) are not significantly impacted at MUR power uprate conditions. Therefore, the extraction steam system is acceptable at power uprate conditions.

VI.1.A.iv Condensate and Main Feedwater Systems

The condensate and feedwater systems are described in UFSAR Section 10.3.5. These systems were evaluated to determine the impact of the MUR power uprate.

VI.1.A.iv.a Condensate System

There are three parallel 50% capacity condensate pumps. Normally two condensate pumps are operating at full load delivering water to the main feedwater pumps suction header. Two low pressure and one high pressure heater drain pumps are normally operating at full load.

The power uprate results in increased condensate flow of approximately 2.1%. Adequate condensate pump net positive suction head is available at uprate conditions. Piping pressures and temperatures are not significantly impacted. Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the condensate system is acceptable at power uprate conditions.

VI.1.A.iv.b Main Feedwater System

There are two parallel motor-driven main feedwater pumps, and both are required for operation at full load conditions. These pumps are constant speed, so feedwater flow is controlled by the feedwater regulating valves on the pump discharge.

The power uprate results in increased feedwater flow of approximately 2%. Adequate main feedwater pump net positive suction head is available at uprate conditions. The increase in extraction steam flow through the feedwater heaters results in a small increase in feedwater temperature entering the SG. Main feedwater isolation valves, feedwater regulating valves, feedwater regulating bypass valves, and main feedwater pump discharge valves provide a containment isolation feature. The existing NSSS accident analysis was completed at 102% of 2546 MWt, which bounds the power uprate. Piping pressures and temperatures are not significantly impacted. Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the main feedwater system is acceptable at power uprate conditions.

VI.1.A.iv.c Abnormal/Transient Operating Conditions

The following transients that impact feedwater flow were evaluated at power uprate conditions: loss of heater drain pump (high pressure or low pressure), loss of a main feedwater pump, and 50% load rejection. There is no significant impact on system operation from any of these postulated transients.

VI.1.A.v Feedwater Heaters

There are two parallel trains of feedwater heaters. Each train consists of five heaters (6th, 5th, 4th, 3rd and 2nd point heaters) located on the suction side of the main feedwater pumps. The 6th and 5th point feedwater heaters are located in the main condenser neck. Two additional feedwater heaters (1st points) are located on the discharge side of the main feedwater pumps.

Relevant feedwater heater parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system functionality. Therefore, the feedwater heaters are acceptable at power uprate conditions.

VI.1.A.vi Feedwater Heater and Moisture Separator Reheater Vents and Drains

The secondary vent and drain systems are described in UFSAR Section 10.3.8. Feedwater heater and moisture separator reheater vents and drains were evaluated at MUR power uprate conditions. Operating parameters (flow, pressure, temperature, velocity) at power uprate conditions do not significantly impact piping, component, and equipment design parameters. Therefore, feedwater heater and moisture separator reheater vents and drains piping, component, and equipment design are acceptable at the MUR power uprate conditions.

VI.1.A.vii Auxiliary Feedwater System

The AFW system design basis of record is described in UFSAR Section 10.3.5. The AFW system serves as a backup system for supplying feedwater to the SGs when the main feedwater system is not available. Each unit's system includes two motor driven pumps and one turbine driven pump configured into two trains. Each train provides feed flow to all SGs. Each pump takes suction through independent lines from the missile protected ECST. The AFW system analyses are based on a core thermal power level of 2597 MWt, which is 102% of 2546 MWt. The analyzed core power level of 2597 MWt remains conservative and bounds the MUR power level. System maximum operating pressure and temperatures remain essentially unchanged as a result of the MUR power uprate. Piping and component pressure and temperatures design parameters bound power uprate operating pressure and temperature conditions. AFW system flow requirements associated with the analysis are bounding for the power uprate. The AFW system has the capacity to provide adequate flow under transient and accident conditions. The current AFW system minimum flow requirements remain acceptable for MUR, and there are no proposed changes to AFW pump design/performance or operation. Since no changes are being made to the pump design, the brake horsepower requirements are unaffected. No AFW system modifications are required to support the MUR power uprate.

The design basis scenario that defines the ECST volume requirement is holding for eight hours in hot shutdown. The minimum required ECST volume is 90,279 gallons. The eight hour integrated decay heat was based on a core power of 2597 MWt, which remains conservative and bounding for the power uprate. The Technical Specification minimum ECST volume requirement of 96,000 gallons ensures that the usable volume bounds the minimum ECST volume requirement. Therefore, the AFW system is acceptable at MUR power uprate conditions.

VI.1.B Containment Systems

The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the EQ acceptance limits.

VI.1.B.i Containment Spray and Recirculation Spray Systems

The CS and RS systems are described in UFSAR Section 6.3.1. They operate to limit peak containment pressure to less than the design pressure of 45 psig during a LOCA or MSLB, to maintain containment structural integrity. Both systems provide a cooling spray into the containment to remove heat from the containment atmosphere. The CS system takes water from the RWST, mixes in sodium hydroxide from the chemical addition tank to assist in iodine removal and to control containment sump water pH, and delivers the discharge through spray rings. The RS system takes water from the containment sump and delivers the discharge through spray rings.

The existing containment response analyses remain bounding for the power uprate. The CS system and RS system operating and design parameters in the existing analyses bound the power uprate parameters. There are no new operating requirements imposed on either system as a result of the power uprate. Therefore, the CS system and RS system are acceptable for operation at MUR uprate conditions.

VI.1.B.ii Containment Air Cooling

The containment ventilation systems are described in UFSAR Section 5.3.1. The containment ventilation system provides general area cooling and direct cooling to critical components. It also provides the means to purge the containment atmosphere prior to personnel entry during maintenance periods. Containment air cooling consists of a recirculation cooling system, CRDM cooling system, filter system, and purge system. The RCP motors are cooled by an integral heat exchanger supplied by the CCW system. The CRDM cooling system is discussed in Section VI.1.B.iii. The recirculation and CRDM cooling systems provide air cooling that in combination with the RCP motor cooling maintain containment bulk air temperature within the Technical Specification limits.

CRDM equipment was analyzed at MUR power uprate conditions. The heat increase to the containment atmosphere from the CRDM system is approximately

250,000 BTU/hr for Unit 1 and approximately 231,000 BTU/hr for Unit 2. NSSS equipment heat load changes were analyzed at MUR power uprate conditions. The heat changes will not affect the containment bulk air temperature. Therefore, the MUR power uprate will have no significant impact on the containment atmosphere.

VI.1.B.iii CRDM Ventilation

The CRDM cooling system was evaluated at power uprate conditions to demonstrate that the electro-magnetic coils design temperature was not exceeded.

Unit 1

The CRDM lift coil temperature after 15 minutes of stepping is the limiting case for maximum coil temperature. At MUR power uprate conditions, the maximum expected electro-magnetic coil temperature after 15 minutes of stepping is 288.3°F. This is below the coil design temperature of 392°F.

Unit 2

The CRDM lift coil temperature after 3.2 minutes of stepping is the limiting case for maximum coil temperature. A previous plant modification reversed the air flow direction from downward to upward. The stepping transient is limited to 15 minutes with downward flowing air (Unit 1), but with upward flowing air the lift coil temperature runs hotter at steady state conditions, so the stepping transient is shortened. The time to fully withdraw or insert a control rod at maximum speed is 3.38 minutes. At power uprate conditions, the maximum expected electro-magnetic coil temperature after 3.2 minutes of stepping is 375.1°F. This is below the coil design temperature of 392°F.

The CRDM coil operating temperatures remain below their design temperature limits at MUR power uprate conditions, without equipment upgrade or changes in operating parameters. Therefore, the Unit 1 and 2 CRDM cooling systems are acceptable at MUR power uprate conditions.

VI.1.C Safety-Related Cooling Water Systems

VI.1.C.i Component Cooling Water System

The CCW system is described in UFSAR Section 9.4. The CCW system is a closed loop piping system shared between Units 1 and 2, and rejects heat to the SW system. There are four CCW pumps and four CCW heat exchangers, which can be cross-connected to share loads between the two units. Normally, two heat exchangers and two pumps (one per unit) are required to support the normal heat loads of both units. The CCW system is designed to provide the cooling requirements for normal plant operation, plant cooldown, and the design basis accident of one unit.

The CCW system was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the MUR power uprate heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. The analysis confirms that at MUR uprated conditions, normal plant operation and required cooldown time continue to be met.

VI.1.C.ii Service Water System

The SW system is described in UFSAR Section 9.9. Water is supplied to each SW system by the CW system which is common to both units. There are four CW pumps per unit. Each CW pump takes suction from the James River and discharges into the CW intake canal. SW flow is provided by gravity feed through valves located upstream of each unit's condenser inlet. The SW system is designed to support a LOCA in one unit, while placing the non-accident unit in a cold shutdown condition in the event of a coincident LOOP. During an accident condition, one or two (depending on the scenario) of three SW pumps are required to provide adequate inventory for heat removal for both units.

Each component cooled by the SW system was evaluated to confirm that the existing flow rate is sufficient to satisfy the power uprate heat removal requirements during normal power operation, accident, and cooldown conditions. The evaluations determined that the existing SW flows will continue to support the heat removal requirements at uprate conditions. The SW system and component design parameters remain bounding for power uprate operation. No system modifications are required to support the power uprate. Therefore, the SW system is acceptable for operation at MUR power uprate conditions.

VI.1.C.iii Ultimate Heat Sink

The ultimate heat sink is comprised of the James River, the CW intake canal, and the discharge canal.

The SW system inlet temperature for normal, cooldown, and DBA conditions is bounded for the power uprate. The ultimate heat sink is capable of cooling the SW system to prevent SW temperature from exceeding the inlet temperature limits during operating conditions. No system modifications are required to support the power uprate. Therefore, the ultimate heat sink is acceptable for operation at power uprate conditions.

VI.1.C.iv Residual Heat Removal System

UFSAR Section 9.3 describes the RHR system. RHR cooldown performance was analyzed under MUR uprate conditions. The normal two train cooldown, one train cooldown, and accident case cooldown were analyzed. The analysis showed that each of these cases met the cooldown time requirements.

VI.1.D Spent Fuel Pool Storage and Cooling Water

The SFP criticality analysis is described in UFSAR Appendix 9A, Section 9A.3.2. UFSAR Section 9.5 describes the SFP cooling and purification system. This system is common to both Surry units.

VI.1.D.i Spent Fuel Pool Criticality

The analysis of record was submitted to the NRC in Reference VI-1 with additional information provided in References VI-2 and VI-3. The NRC approved the analysis in Reference VI-4.

Dominion performed an evaluation to determine the MUR power uprate impact on the SFP criticality analysis of record which considers the SFP as two distinct regions. The power uprate has no effect on the fresh fuel characteristics, so this portion of the analysis (Region 2) is unaffected. For irradiated fuel (Region 1), the cask drop accident analysis is potentially affected. MUR has no effect on optimum pin pitch, the major source of conservatism in the analysis. The MUR also has a negligibly small indirect effect on the calculation of conservative depleted fuel isotopic concentrations due to possible soluble boron increases. The primary effect of the MUR is a small increase in fuel depletion power; however, neither the analysis nor licensing basis require a specific value or degree of conservatism for the depletion power. The Region 1 burnup credit analysis also includes excess identifiable margin. Based on these considerations, the SFP criticality analysis will remain applicable for fuel used in MUR cycles.

VI.1.D.ii Spent Fuel Pool Cooling and Purification

SFP cooling heat exchangers are cooled by CCW. Heat exchanger outlet flow returns to the SFP or is sent to the refueling purification system, consisting of an ion exchanger and filter.

There are no changes to the SFP cooling system limiting temperatures, pressures or flow rates as a result of the power uprate. Uprate conditions are bounded by the existing system design conditions. System modifications are not required to support the power uprate. The limiting case heat loads at uprate conditions remain bounded by the existing analysis. There is no change to the loss of cooling analysis. The power uprate is not expected to have any significant impact on the SFP refueling purification or cooling functions. Therefore, the SFP cooling and purification system is acceptable at the MUR power uprate conditions.

VI.1.E Radioactive Waste Systems

VI.1.E.i Gaseous Waste

The gaseous waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive gases released during simultaneous operation of both units. Gaseous waste system functions and the volume of waste gas processed are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The gaseous waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.ii Liquid Waste

The liquid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive liquid waste produced during simultaneous operation of both units. Liquid waste system functions and the liquid waste volume processed are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The liquid waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.iii Solid Waste

The solid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units and is sized to treat the radioactive solid waste produced during simultaneous operation of both units. Solid waste system functions and the volume of solid waste volume processed are unaffected by the uprate. No system or component design parameters were exceeded at uprate conditions. The solid waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.iv Steam Generator Blowdown

The required SG blowdown flow rates during plant operation are based on chemistry control and tubesheet sweep necessary to control solids buildup. The SG blowdown system was analyzed for a blowdown flowrate increase of approximately 4 gpm. However, SPS will continue to operate the SG blowdown system per the plant chemistry program with no change in blowdown flowrate attributable to the power uprate. Blowdown system operating temperatures and pressures will decrease and remain bounded by the existing design parameters under uprate conditions.

The uprate will not significantly increase the potential for FAC on the blowdown system piping and components. SPS will continue to monitor the blowdown system for FAC. Therefore, the SG blowdown system will continue to meet system design requirements at MUR power uprate conditions.

VI.1.F Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning

VI.1.F.i Control Room Ventilation System

UFSAR Section 9.13.3.6 describes the main control room, emergency switchgear and relay room ventilation systems. The main control room and emergency switchgear rooms envelope has two independent air conditioning systems consisting of two air handling units (one for the main control room and one for the emergency switchgear room), chilled water piping and a water chiller (one chiller for one loop and two chillers for two-loop operation). The main control room/emergency switchgear room chilled water systems are independent of the station chilled water system. The main control and computer room air conditioning is designed to maintain 75°F and approximately 50% relative humidity in the associated rooms during either normal or emergency conditions. The relay rooms are designed for 80°F dry bulb during normal conditions and 87°F dry bulb during emergency operations.

The heat loads (electrical heat loads, lighting, personnel) at MUR power uprate conditions were evaluated. Radiological consequences of the MUR are discussed in Section II.2.24. The main control room and computer room, relay rooms, emergency switchgear room normal and emergency ventilation systems, and chilled water systems are not impacted by the MUR power uprate conditions, because the heat loads in these areas do not increase.

VI.1.F.ii ESF Ventilation System

The safeguards area ventilation system is described in UFSAR Section 9.13.3.4. A separate safeguards area ventilation system is provided for Units 1 and 2. The safeguards area ventilation system is designed to limit temperatures to 120°F during warm weather and to raise incoming outside air to a minimum temperature of 50°F during cold weather.

The current limiting case heat loads have been evaluated at the MUR power uprate conditions. The safeguards area ventilation capabilities are not impacted, because there is no increase in electrical heat loads and no significant increase in piping system heat loads at the MUR power uprate conditions. There is a small heat load increase from the feedwater piping and a small heat load decrease from the main steam piping resulting in no significant impact to MSVH ambient air temperature.

VI.1.F.iii Fuel Handling Area Ventilation System

The fuel building ventilation system is described in UFSAR Section 9.13.3.2. The fuel building ventilation system is a once through ventilation system that provides the fuel building with 100% outdoor air, after it has been filtered and heated as required. The ventilation system consists of two supply fans, one that serves the SFP area and one for the remote equipment space. Both supply fans take suction from a common plenum fitted with a combination roll and high efficiency filters and steam coils for air tempering and space heating. The ventilation system maintains a maximum air temperature of 105°F and a minimum air temperature of 75°F.

The SFP cooling equipment loads analyses are not impacted by the MUR power uprate. As discussed in Section II.2.32, the higher decay heat loads will not impact the limiting case full core off-load. The maximum SFP and piping temperatures at MUR conditions will be at or below the calculated limiting case. The fuel building ventilation system is not impacted by the MUR power uprate, because there is no increase in the SFP temperature, piping or electrical heat loads.

VI REFERENCES

VI-1 Letter from James P. O'Hanlon (Virginia Electric and Power Company) to USNRC Document Control Desk, *Virginia Electric and Power Company, Surry Power Station Units No. 1 and 2, Proposed Technical Specifications Change for Increased Enrichment of Reload Fuel,* Serial No. 97-614, November 5, 1997.

VI-2 Letter from James P. O'Hanlon (Virginia Electric and Power Company) to USNRC Document Control Desk, *Virginia Electric and Power Company, Surry Power Station Units No. 1 and 2, Increased Fuel Enrichment Technical Specifications Change Response to NRC Request for Additional Information,* Serial No. 98-010, January 28, 1998.

VI-3 Letter from James P. O'Hanlon (Virginia Electric and Power Company) to USNRC Document Control Desk, *Virginia Electric and Power Company, Surry Power Station Units No. 1 and 2, Increased Fuel Enrichment Technical Specifications Change Response to NRC Request for Additional Information,* Serial No. 98-237, May 12, 1998.

VI-4 Letter from Gordon E. Edison (USNRC) to J.P. O'Hanlon (Virginia Electric and Power Company), *Surry Units 1 and 2 – Issuance of Amendments Re: Increased Enrichment of Reload Fuel (TAC Nos. MA0122 and MA0123),* ML01270055, June 19, 1998.

VII. OTHER

1. A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.

2. A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

   A. emergency and abnormal operating procedures.

   B. control room controls, displays (including the safety parameter display system) and alarms.

   C. the control room plant reference simulator.

   D. the operator training program.

3. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.

4. A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2% to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

5. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

   A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

   B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

RESPONSE TO VII - OTHER

VII.1 Operator Actions

Operator actions included in the safety analyses were reviewed for potential MUR power uprate impact. The following design basis events were reviewed:

Appendix R Fire	UFSAR Section 9.10
Boron Dilution	UFSAR Section 14.2.5
SGTR	UFSAR Section 14.3.1
Rupture of Main Steam Pipe	UFSAR Section 14.3.2
Fuel Handling Accident	UFSAR Section 14.4.1
VCT Rupture	UFSAR Section 14.4.2.1
Large Break LOCA	UFSAR Section 14.5.1
SBLOCA	UFSAR Section 14.5.2
Loss of Normal Feedwater	UFSAR Section 14B.6

The safety analysis reviews have determined that the existing required operator actions are not affected by the MUR power uprate. There is no reduction in time for required operator actions. No new manual operator actions were created and no existing manual actions were automated.

The power uprate is being implemented under the administrative controls of the plant modification process. Other potential impacts on operator actions and action times in plant procedures may be identified and evaluated during the plant modification impacts review. The plant modification process ensures that impacted procedures will be revised prior to the power uprate implementation.

VII.2.A Emergency and Abnormal Operating Procedures

Emergency and abnormal operating procedures were reviewed to determine any MUR power uprate impact. No changes are required to the procedure steps and mitigation actions as a result of the MUR power uprate. Setpoints used in the plant emergency and abnormal procedures were based on a core thermal power of 2546 MWt. The use of core rated thermal power in developing the setpoints was consistent with the Westinghouse Owners Group background documents. For the implementation of the MUR power uprate, the setpoints will be changed to reflect an MUR power of 2587 MWt.

There are no operator action changes for shutdown risk management due to MUR power uprate. The time to core boiling will decrease due to the MUR but the method of calculating the time to boil will remain the same. SPS procedures will be revised with data generated with decay heats at the MUR power level.

Operator training on the procedure changes will be provided as part of the MUR implementation.

VII.2.B Control Room Controls, Displays and Alarms

The following changes/modifications associated with the proposed power uprate affect control room controls:

- Instruments associated with turbine first stage pressure will require scaling changes for NSSS protection permissives P-2 and P-7, AMSAC input, and high steam flow, steam dump control, steam generator level control and rod control.

The following modifications associated with the proposed power uprate affect operator displays (including the safety parameter display system):

- Instrument loops are affected by the power uprate (indicator replacement, calibration span, and/or scaling).
- Plant computer points will be added and/or changed for the revised calorimetric algorithm and the feedwater UFM.
- No significant safety parameter display system changes are anticipated as a result of the MUR power uprate. Critical safety function status trees will be reviewed and revised as necessary.
- The new UFM electronic cabinet, located in the Cable Spreading Room, is used to display and control aspects of feedwater flow data. The display provides system status or monitored process parameters. The display is typically used for maintenance purposes and not for control of plant operations.

The following modifications associated with the proposed power uprate affect alarms:

- The system alerts operations personnel of UFM trouble through main control room overhead annunciator "Feedwater Ultrasonic Flow Trouble." The main control room overhead annunciator "Feedwater Ultrasonic Flow Meter Failure" alerts the operators when the system loses a plane of operation, suffers a loss of AC power or other total failure. Any UFM condition that increases feedwater flow uncertainty is considered a "Feedwater Ultrasonic Flow Meter Failure" alarm condition.

VII.2.C Control Room Plant Reference Simulator

The MUR power uprate is being implemented under the plant modification process administrative controls. As part of this process, potential simulator modifications will be identified. Simulator required changes resulting from the MUR power uprate will be evaluated, implemented and tested per SPS approved procedures. Simulator fidelity will be revalidated per SPS approved procedures. Any required simulator modifications will be completed in time to support operator training prior to MUR power uprate implementation.

VII.2.D Operator Training Program

The operator training program requires revision as a result of the MUR power uprate. Operator training will be developed and the operations staff will be trained on the plant modifications, Technical Specification and TRM changes, and procedure changes prior to MUR power uprate implementation.

VII.3 Intent To Complete Modifications

Dominion will complete the modifications required to support the MUR (including operator training) prior to power uprate implementation.

VII.4 Temporary Operation Above Licensed Power Level

Dominion will revise the existing plant operating procedure related to temporary operation above full steady-state licensed power levels. Precautions will be revised to account for the uprate power level.

VII.5 10 CFR 51.22 Discussion

VII.5.A 10 CFR 51.22 provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (A.1) involve a significant hazards consideration, (A.2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (B) result in a significant increase in individual or cumulative occupational radiation exposure.

Dominion has determined that this license amendment request meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment is required in connection with issuance of the proposed license amendment. The basis for this determination follows:

1. The proposed license amendment does not involve a significant hazards consideration as previously described in Attachment 1, Section 5.0 Regulatory Analysis for this License Amendment Request.

2. The proposed change does not involve installing new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite.

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite during normal operation. The primary coolant specific activity is expected to increase by no more than the percentage increase in power level.

Gaseous and liquid radwaste effluent activity is expected to increase from current values by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and 10 CFR 50, Appendix I limits per the Surry Offsite Dose Calculation Manual. The proposed changes will not result in changes to the operation or design of the gaseous or liquid waste systems and will not create any new or different radiological release pathways.

Solid radwaste effluent activity is expected to increase from current levels proportionately to the increase in long half-life coolant activity. The total long lived activity is bounded by the percent of power uprate. Changes in solid waste volume are not expected.

Therefore, the proposed license amendment request will not result in a significant change in the types or significant increase in the amounts of effluents that may be released offsite.

VII.5.B The proposed license amendment request does not significantly increase core power and resultant dose rates in accessible plant areas. Normal operation radiation levels will increase by approximately the percentage of core power uprate. The power uprate does not require additional radiation shielding to support normal plant operation. Individual worker exposures will be maintained within acceptable limits by the site Radiation Protection Program, which controls access to radiation areas and maintains compliance with 10 CFR 20.

Therefore, the proposed license amendment request does not result in a significant increase to the individual or cumulative occupational radiation exposure.

VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

UFSAR Section 9.10 describes the SPS Fire Protection Program. The Fire Protection Program satisfies the regulatory criterion of General Design Criteria 3; 10 CFR 50, Appendix R (Sections III.G, III.J, III.L, and III.O); and Branch Technical Position APCSB 9.5-1.

VII.6.A.i Fire Protection Systems

The Fire Protection System consists of the following major subsystems: fire detection (including smoke detectors, heat detectors, alarms), water suppression (including fire pumps, main fire loop piping, sprinkler systems, deluge systems), CO_2 suppression, Halon suppression, manual fire equipment (portable fire extinguishing equipment), and fire barriers (including fire walls, fire doors, penetration seals, cable wraps, cable tray stops, heat shields). The fire protection subsystems remain unchanged as a result of the MUR power uprate.

The Surry Fire Protection System is not credited or required to mitigate the consequences of Design Basis Accidents. However as noted in UFSAR Section 9.10.1, in addition to its primary function, which is to permit safe shutdown of the plant in the event of a fire, the fire protection system also provides alternate sources of makeup water for the spent fuel pool, for the Unit 1 and Unit 2 auxiliary feedwater systems, and for backup water source to the Unit 1 and Unit 2 bearing cooling system for cooling the instrument air compressors. In accordance with BTP-APCSB 9.5-1, Appendix A, Paragraph A.4, postulated fires need not be considered concurrently with other plant accidents. Therefore, these secondary functions of the fire protection system do not prohibit the system from performing its primary function. Additionally, the fire protection system's capacity remains adequate to provide backup water to the auxiliary feedwater pumps, makeup water to the spent fuel pool, and a backup source to the bearing cooling system for cooling the instrument air compressors at the uprate power conditions.

VII.6.A.ii Responsibilities

Plant management, supervisory and station personnel responsibilities in support of the Fire Protection Program are not impacted by the MUR power uprate.

VII.6.A.iii Administrative Controls

Topics include control and use of fire protection systems and equipment; combustibles storage; control of ignition sources; implementing ventilation for heat and smoke removal; design change control for fire protection systems and equipment; Fire Protection Program instructions, procedures, and drawings; fire inspection program; fire equipment maintenance and testing; and fire strategies. The MUR power uprate does not affect the established administrative controls.

VII.6.A.iv Fire Brigade

There are no changes in the fire brigade structure, responsibilities, reporting relationships, or qualifications resulting from the MUR power uprate.

VII.6.A.v Evaluations of Inadvertent Operation of Fire Protection Systems

The MUR power uprate does not affect the existing evaluation conclusions for the inadvertent operation of fire protection systems.

VII.6.B High Energy Line Break Program

The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. UFSAR Appendix 14B, "Effects of Piping System Breaks Outside Containment," describes the high and moderate energy line break analysis. High-energy pipe breaks are analyzed for piping where the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200°F. Cracks are postulated in the moderate energy piping where either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200°F.

The evaluation concluded that the MUR power uprate does not result in any new or revised high or moderate energy line break locations. The high and moderate energy line break analysis is not affected. Area temperature and pressure resulting from HELBs and internal flooding conditions resulting from moderate energy line breaks for the buildings remain valid at MUR power uprate conditions.

VII.6.C Appendix J Program

UFSAR Section 5.5, "Containment Tests and Inspections," states that a performance based testing program will include Type A tests to measure the containment overall integrated leakage rate, Type B tests to detect and measure local leakage pressure-containing or leakage-limiting boundaries other than valves, and Type C tests to measure containment isolation valve leakage rates. The containment leakage tests are performed as required by 10 CFR 50, Appendix J, Option B.

A review of the LOCA response analysis confirmed that the analysis was performed at 102% of 2546 MWt. Because the LOCA peak pressure analysis is unaffected, P_a at MUR power uprate conditions is unchanged from the current conditions specified in SPS Technical Specification 4.4. No changes or modifications are required to the existing Appendix J Program or procedures. Therefore, SPS Technical Specification 4.4 and the applicable SPS Appendix J Program procedures are acceptable at MUR uprate conditions.

VII.6.D Coatings Program

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and radionuclide contamination. Coatings also provide wear protection during plant operation and maintenance activities. These coatings are subject to 10 CFR 50, Appendix B quality assurance requirements, since their degradation could adversely impact safety related equipment. The approved SPS containment Service Level 1 coatings are qualified to withstand a LOCA environment and meet ANSI Standards N5.12, N101.2 and N101.4.

The UFSAR LOCA containment response analyses remain bounding for the MUR power uprate. There were no changes to the containment analyses that would require a change to the containment design pressure or temperature. Since the containment design pressure and temperature limits were used to qualify the Service Level 1 containment coatings, and those limits are not changing, the Service Level 1 containment coatings remain qualified under MUR power uprate conditions.

VII.6.E NRC Generic Letters

The design criteria for safety-related valves are promulgated in 10 CFR 50.55a. Additional information is also provided by the plant specific evaluations of GL 89-10, GL 95-07 and GL 96-06. The plant specific provisions of GL 89-10, GL 95-07, and GL 96-06 were reviewed to determine if any changes were required as a result of the power uprate. No required changes were identified.

VII.6.E.i GL 89-10 Motor Operated Valve (MOV) Program

The NRC issued GL 89-10 (Reference VII-1) requiring licensees to develop a comprehensive program to ensure MOVs in safety-related systems would operate under design basis conditions.

The review determined that the maximum differential pressures/line pressures determined in the system and functional design basis review calculations for the GL 89-10 identified MOVs were not affected by the MUR power uprate. The values for these parameters at current conditions bound the values at MUR conditions. Therefore, these parameters do not affect the calculations that determine MOV thrust and torque values. The MOV flow rates documented in the system and functional design basis review calculations for the GL identified MOVs at current conditions bound the flow rates at MUR conditions. The power uprate does not affect the maximum ambient temperatures used to determine MOV motor capability torque values at current conditions. The MUR power uprate has no effect on valve factors or required thrusts because pressure, temperature and flow conditions are not a direct input into calculating the valve factor. Therefore the conclusions previously provided for GL 89-10 identified MOVs are not impacted by the MUR power uprate.

VII.6.E.ii GL 95-07 Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves

The NRC issued GL 95-07 (Reference VII-2) to address potential pressure locking and thermal binding of safety-related power operated gate valves. SPS responded to this GL in References VII-4, VII-5, VII-6, and VII-7.

The review determined that the MUR power uprate does not affect the pressure locking evaluations previously completed. The thrust required to open the applicable valves remains less than the motor actuator capabilities at power uprate conditions. The power uprate does not affect valve design, valve function, or operational conditions. New conditions were not created that would affect valve susceptibility to pressure locking or thermal binding. Therefore, the conclusions previously provided in References VII-4 through VII-7 for valve pressure locking and thermal binding acceptability are not impacted by the MUR power uprate.

VII.6.E.iii GL 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The NRC issued GL 96-06 (Reference VII-3) to address hydrodynamic effects of water hammer and two-phase flow conditions on cooling systems serving containment air coolers and thermally induced overpressurization of isolated piping segments. SPS responded to this GL in References VII-8, VII-9, VII-10, VII-11, VII-12, and VII-13.

Containment air cooling system two-phase flow and water hammer are not applicable, because the system is isolated and de-energized during design basis accidents. The MUR power uprate does not modify system configuration or change system operation. The CS and RS systems piping is not filled with water until after a containment depressurization actuation signal. Thus, system piping overpressurization cannot occur prior to system actuation. The CS and RS systems are not modified and system operating parameters are unchanged. The current LOCA accident analyses were performed at 102% of 2546 MWt and remain bounding for the MUR power uprate. There is no increase in the possibility of overpressurizing isolated segments of safety-related piping inside containment, including penetrations, as a result of the power uprate. Therefore, there is no impact regarding GL 96-06 program issues at MUR power uprate conditions.

VII.6.F Air Operated Valve Program

The SPS AOV program includes the following categories of AOVs:

Category 1 – AOVs that are high safety significant

Category 2 – AOVs that are low safety significant, safety-related, and non-safety related with special regulatory significance

The system evaluations for Category 1 AOVs indicate that the MUR does not affect the maximum differential pressures/line pressures, flow rates, or fluid temperatures documented in the system level design basis review calculations. Therefore, the MUR power uprate does not affect the AOV setup values in the component level calculations for these AOVs.

The system evaluations for Category 2 AOVs indicate that the existing maximum operating flow rates and pressures are bounded by the current parameters and remain valid at MUR power uprate conditions.

VII REFERENCES

VII-1 NRC Generic Letter 89-10, *Safety-Related Motor Operated Valve Testing and Surveillance*, June 28, 1989 and supplements.

VII-2 NRC Generic Letter 95-07, *Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves*, August 17, 1995.

VII-3 NRC Generic Letter 96-06, *Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,* September 30, 1996.

VII-4 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,* Serial No. 95-566, November 15, 1995.

VII-5 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,* Serial No. 95-566A, February 7, 1996.

VII-6 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, Request for Additional Information,* Serial No. 96-315, July 3, 1996.

VII-7 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, Request for Additional Information (RAI),* Serial No. 99-333, August 6, 1999.

VII-8 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, NRC Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,* Serial No. 96-516, October 30, 1996.

VII-9 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, NRC Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,* Serial No. 96-516A, January 28, 1997.

VII-10 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Supplemental Response to Generic Letter (GL) 96-06,* Serial No. 96-516B, October 23, 1997.

VII-11 Letter from James P. O'Hanlon (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Supplemental Response to Generic Letter (GL) 96-06, Acceptance Criteria for Design Adequacy Evaluation,* Serial No. 96-516C, February 25, 1998.

VII-12 Letter from D. A. Christian (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry and North Anna Power Stations Units 1 and 2, Supplemental Response to Generic Letter (GL) 96-06, Structural Integrity Evaluation of Thermally Induced Overpressurization of Containment Penetration Piping During DBA,* Serial No. 99-134, March 30, 1999.

VII-13 Letter from D. A. Christian (Virginia Power) to USNRC, *Virginia Electric and Power Company, Surry and North Anna Power Stations Units 1 and 2, Generic Letter (GL) 96-06, Response to Request for Additional Information,* Serial No. 99-320, June 22, 1999.

VIII. CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

   A. a description of the change

   B. identification of analyses affected by and/or supporting the change

   C. justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change

RESPONSE TO VIII - CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

VIII.1 Technical Specification Changes

VIII.1.A Description of Change

Table VIII-1
Description of Technical Specifications Changes

Change No.	Change Description
1	**Facility Operating License, Paragraph 3.A** Dominion is authorized to operate the facility at a reactor core power levels not in excess of 2587 megawatts (thermal)
2	**TS Section 1.0, Definitions – RATED POWER** RATED POWER shall be a steady state reactor core heat output of 2587 MWt.
3	**TS Figure 2.1-1 – Reactor Core Thermal and Hydraulic Safety Limits Three Loop Operation, 100% Flow** Revised reactor core safety limit lines reflect MUR operating conditions at 2587 MWt.
4	**TS Section 2.3.A.2(d) – Overtemperature ΔT** The OTΔT pressure adjustment term, K_3, is determined to equal 0.000770.

The following information provides the supporting justification for the proposed Technical Specifications changes described above.

VIII.1.B Supporting Analysis

The current SPS RP is 2546 MWt. The MUR power uprate will increase power by approximately 1.6%. This increase is based on a plant specific evaluation of reactor power measurement uncertainty using the UFM instrumentation versus the 10 CFR 50, Appendix K previously mandated 2% uncertainty. Therefore, the new RP will be:

RP = 2546 MWt * 1.0161 = 2587 MWt

VIII.1.C Justification for Changes

Detailed evaluations and analyses were performed demonstrating that SPS operation at a reactor power level of 2587 MWt is acceptable. The detailed evaluations and analyses considered the effects of operation at this power level on: power level measurement uncertainty; postulated accidents and transients; mechanical, structural and material components integrity and design; electrical equipment design; system design; operator actions, emergency and abnormal operating procedures, control room, plant simulator, and operator training; environmental impact; and Technical Specifications, protection system settings, and emergency system settings.

The evaluations and analyses were performed using current licensing basis acceptance criteria and Technical Specifications. This ensures the same protection level for public health and safety at the uprated conditions as the currently licensed power level. These evaluations and analyses are described in this attachment. Attachment 2 contains the Operating License and Technical Specification marked-up pages and Attachment 3 contains the typed pages to reflect the proposed changes.

A comprehensive review of the Reactor Core Safety Limits (RCSLs), reactor protection system (RPS) setpoints, Engineered Safety Features Actuation System (ESFAS) setpoints, and the Technical Specifications (TS) was performed to evaluate the impact of the proposed core power uprate to 2587 MWt. The results of this review are provided below.

The impact of the proposed uprate on reactor core safety limit lines (TS 2.1) and on the associated thermal overpower ΔT (OPΔT) and overtemperature ΔT (OTΔT) reactor protection setpoints was evaluated.

- TS 2.1 identifies the Reactor Core Safety Limits, which provide the combination of reactor thermal power level, coolant pressure and coolant temperature that ensure fuel limits are met. Revised core safety limit lines for Technical Specifications Figure 2.1-1 were developed to reflect MUR operating conditions at 2587 MWt. Figure VIII-1 documents the limit lines that were generated using the NRC-approved methodology in Reference VIII-1. Figure VIII-1 replaces current TS Figure 2.1-1. TS Figures 2.1-2 and 2.1-3 provide limit lines for N-1 loop operation with loop stop valves open and closed, respectively. N-1 loop power operation is not permitted for Surry, so no changes to these figures are proposed. A separate licensing submittal proposes to eliminate TS Figures 2.1-2, 2.1-3, and 2.1-4 (Reference VIII-2).

- The overtemperature/overpower ΔT protection function was reviewed for the MUR power uprate and associated plant conditions in accordance with the methodology in Reference VIII-1. A change to the OTΔT trip pressure constant (K3) from 0.000566 to 0.000770 is required to ensure protection at low RCS pressures (i.e., down to the low pressure reactor trip Technical Specification 2.3.A.2(c) setting limit of 1875 psig). The change to the K3 constant requires a change to TS 2.3.A.2(d). The current TS 2.3 OTΔT setpoint equation (provided below) is not modified. Only the K3 constant will be changed.

$$\Delta T \leq \Delta T_0\left[K_1 - K_2\left(\frac{1+t_1 s}{1+t_2 s}\right)(T-T') + K_3(P-P') - f(\Delta I)\right]$$

where

ΔT_0 = Indicated ΔT at rated thermal power, °F
T = Average coolant temperature, °F
T' = 573.0°F
P = Pressurizer pressure, psig
P' = 2235 psig
K_1 = 1.135
K_2 = 0.01072
K_3 = 0.000770 (revised from 0.00566)
ΔI = qt – qb, where qt and qb are the percent power in the top and bottom halves of the core respectively, and qt + qb is total core power in percent of rated power
$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Technical Specifications Figure 2.3-1
t_1 $\geq$ 29.7 seconds
t_2 $\leq$ 4.4 seconds

- In accordance with WCAP-8745-P-A (Reference VIII-1), confirmation of protection at MUR conditions with the revised OTΔT K3 pressure constant has been demonstrated. Specifically, a reanalysis of the limiting OTΔT event, Uncontrolled Rod Withdrawal at Power in UFSAR Section 14.2.2, was performed using Dominion's NRC approved transient analysis methodology (Reference VIII-3) and DNBR analysis methodologies (References VIII-4 through VIII-6). This analysis was performed consistent with the restrictions and limitations of the NRC's Safety Evaluation Reports (SERs) for these methods. The SERs are incorporated into the NRC-approved versions of the topical reports. These methodologies are already identified in Surry UFSAR Section 14.2.2 for the RWAP event.

- A series of cases was examined at varying initial power levels and reactivity insertion rates to define the limiting analysis case and to confirm that the OTΔT and high neutron flux trips continue to provide robust protection against violation of the statistical DNBR design limit over the full range of potential system conditions for the MUR uprate. The limiting single failure for this event remains the failure of a single channel of reactor protection.

- DNBR analyses were performed at initial power levels of 101.7%, 60% and 10% of 2546 MWt (current rated thermal power). For each analyzed initial power level, nominal values of the initial RCS average temperature and pressure were assumed, consistent with Dominion's Statistical DNBR Evaluation Methodology (Reference VIII-4). The Technical Specifications 3.12.F minimum measured RCS flow rate of 273,000 gpm was assumed consistent with the analysis of record.

- The high neutron flux reactor trip setpoint was assumed to be 118% of 2589.3 MWt (101.7% of the current rated thermal power of 2546 MWt) for all cases.

- The OTΔT K3 constant was input as 0.000770.

- For a given initial power level, the limiting DNBR occurs for the reactivity insertion rate that produces an OTΔT and high neutron flux trip at essentially the same time. Figure VIII-2 shows the results of the MUR analysis at 101.7% power. The intersection of the different protection functions compares closely to UFSAR Figure 14.2-5. For this reanalysis, a minimum DNBR of 1.64 occurred for an initial condition of 101.7% of 2546 MWt, minimum reactivity feedback, and a reactivity insertion rate of 0.4 pcm/second. The combined effect of the OTΔT K3 increase and the power increase results in a slightly lower reactivity insertion rate becoming the limiting case (UFSAR limiting case is at 0.8 pcm/second). The minimum DNBR of 1.64 is a small decrease compared to the current limiting result of 1.68 at 2546 MWt. The new analysis minimum DNBR is above the DNBR design limit of 1.46. Further, a penalty against retained DNBR margin for the MUR power uprate is not required for the RWAP event in UFSAR Section 14.2.2.

- Analyses to verify that the maximum pressure in the RCS and main steam system (MSS) do not exceed applicable limits (110% of design pressure) were performed using initial conditions that are consistent with the analysis of record. The analysis shows that the RCS and MSS pressure relieving devices continue to have sufficient capacities to ensure the safety of the unit without relying on the mitigating capabilities of the pressurizer pressure control or main steam bypass systems. The current UFSAR identifies the peak RCS pressure as

2742.34 psia, which is from a case with filled pressurizer safety valve (PSV) loop seals. Currently, Surry operates with the PSV loop seals drained. For this configuration, the new RWAP analysis produces a peak RCS pressure of 2699.0 psia for the case at 12% power with a reactivity insertion rate of 55 pcm/second that trips on high neutron flux (118% of 2589.3 MWt). This is the same limiting statepoint identified in UFSAR Section 14.2.2. A base case run with PSV loop seals drained and high neutron flux at 118% of 2546 MWt produced a peak RCS pressure of 2697.2 psia. Thus, the change in the assumed high neutron flux trip setpoint alone increases the peak RCS pressure by less than 2 psi. The analysis result is less than the 2750 psia limit. Adequate RCS overpressure protection is demonstrated.

- For the MSS overpressure analyses, the limiting case occurs at 60% power with a maximum system pressure of 1190 psia, which is less than the limit of 1210 psia. The limiting case is the same as the current UFSAR. Adequate MSS overpressure protection is demonstrated.
- None of the RWAP cases resulted in a pressurizer overfill condition. The limiting case had 73 ft^3 of margin to overfill, compared to 79 ft^3 in the current analysis.
- As part of the MUR uprate implementation, UFSAR Section 14.2.2 will be updated to reflect the RWAP reanalysis.
- In conclusion, the revised RWAP analysis incorporated a bounding MUR power level of 2589.3 MWt (101.7% of 2546 MWt) and a change to the OTΔT K3 constant to 0.000770. The analysis met all UFSAR acceptance criteria. In addition, an evaluation of the proposed OTΔT K3 pressure constant change was performed on plant operating margins, and the margin to trip was found to be acceptable. Therefore, the proposed change to the OTΔT K3 pressure constant is acceptable.

Figure VIII-1
Reactor Core Thermal and Hydraulic Safety Limits
Three Loop Operation, 100% Flow

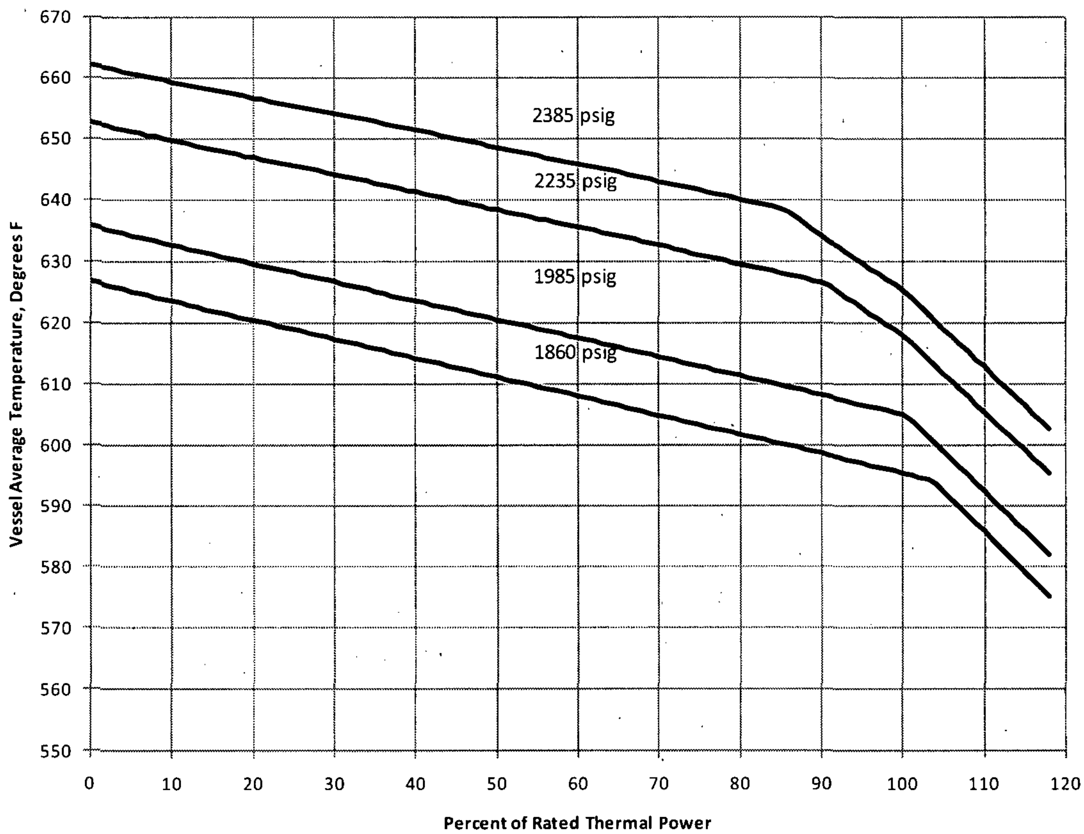
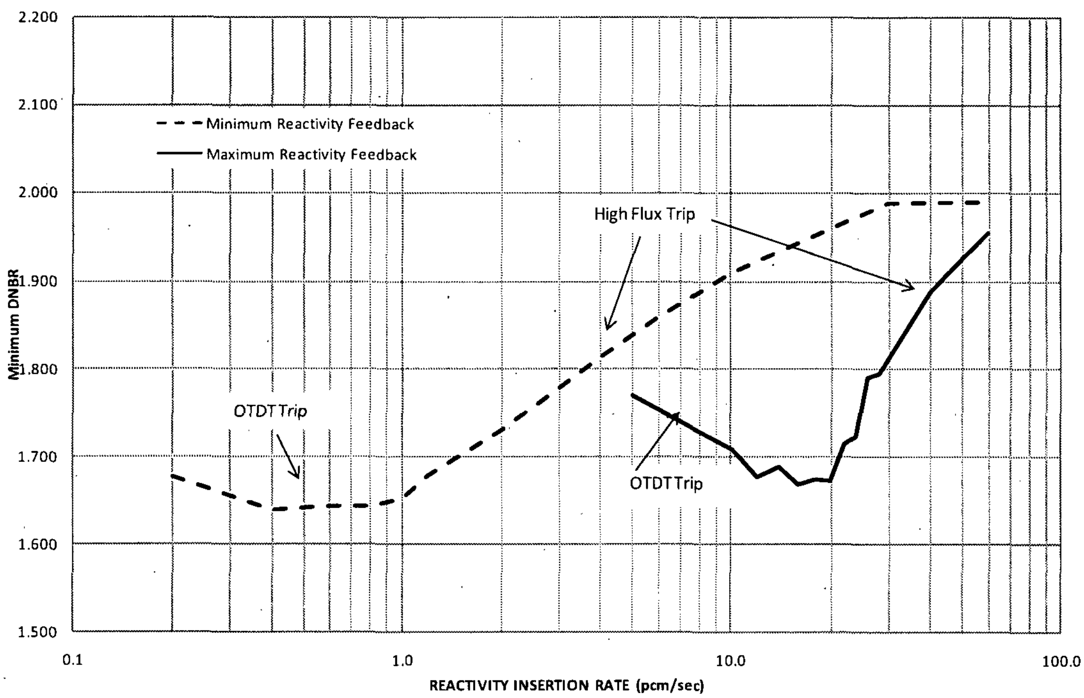


Figure VIII-2
Surry Rod Withdrawal at Power
Minimum DNBR vs. Insertion at 2589.3 MWt (101.7% of 2546 MWt)



VIII.2 Protection System Settings Changes

All other RPS setpoints as defined in TS Section 2.3, Limiting Safety System Settings, Protective Instrumentation, remain acceptable for the proposed MUR uprate and will not changed.

VIII.3 Emergency System Settings Changes

The ESFAS functions and associated setpoints in TS Table 3.7-4, Engineered Safety Feature System Initiation Limits Instrument Setting, remain acceptable for the MUR power uprate and will not change.

VIII REFERENCES

VIII-1 WCAP-8745-P-A, *Design Basis for the Thermal Overpower Delta-T and Overtemperature Delta-T Trip Functions,* September 1986.

VIII-2 Letter from J. Alan Price (Dominion) to USNRC, *Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed License Amendment Request, Relocation of Core Operating Limits to the Core Operating Limits Report (COLR) and Addition of COLR References,* Serial No. 09-581, October 16, 2009.

VIII-3 VEP-FRD-41, Rev. 0.1-A, *VEPCO Reactor System Transient Analysis Using the RETRAN Computer Code,* June 2004.

VIII-4 VEP-NE-2-A, *Statistical DNBR Evaluation Methodology,* June 1987.

VIII-5 VEP-FRD-33-A, Revision 0, *Reactor Core Thermal-Hydraulic Analysis Using The COBRA IIIC/MIT Computer Code,* October 1983.

VIII-6 VEP-NE-3-A, *Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code,* July 1990.

ATTACHMENT 6

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

LIST OF REGULATORY COMMITMENTS

SURRY POWER STATION UNITS 1 & 2
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

REGULATORY COMMITMENTS

The following list identifies those actions committed to by SPS in this LAR. Any other actions discussed in the submittal represent intended or planned actions described for information only and are not regulatory commitments.

COMMITMENT	SCHEDULED COMPLETION DATE (if required)
1. Dominion will perform the final acceptance of the Surry 2 uncertainty analysis to ensure the results are bounded by the statements contained in this LAR (Attachment 5 Section I.1.D.4.1).	Prior to operating above 2546 MWt (98.4% RP).
2. Technical Requirements Manual (TRM) will be revised to include UFM administrative controls (Attachment 1 Section 3.0).	Prior to operating above 2546 MWt (98.4% RP).
3. Revise procedures, programs, and documents for the new UFM (including transducer replacement) (Attachment 5 Section I.1, I.1.D.1.1, I.1.H, VII.1, VII.2.A, and VII.4).	Prior to operating above 2546 MWt (98.4% RP).
4. Appropriate personnel will receive training on the UFM and affected procedures (Attachment 5 Sections I.1.D.1.1, VII.2.A, VII.2.D, and VII.3).	Prior to operating above 2546 MWt (98.4% RP).
5. The FAC CHECWORKS SFA models will be updated to reflect the MUR power uprate conditions (Attachment 5 Section IV.1.E.iii).	Prior to operating above 2546 MWt (98.4% RP)
6. Simulator changes and validation will be completed (Attachment 5 Section VII.2.C).	Prior to operating above 2546 MWt (98.4% RP).
7. Revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels (Attachment 5 Section VII.4).	Prior to operating above 2546 MWt (98.4% RP).
8. Process UFSAR changes in accordance with 10 CFR 50.59 (Attachment 1, Section 3.0).	In accordance with 10 CFR 50.71(e).
9. UFM commissioning and calibration will be completed (Attachment 5, Section I.1.D.2.1).	April 2010
10. Confirm flow normalization factors (Attachment 5, Section I.1.G).	Prior to operating above 2546 MWt (98.4% RP).

COMMITMENT	SCHEDULED COMPLETION DATE (if required)
11. Rescaling and calibration of main turbine first stage pressure input to AMSAC (Attachment 5, Sections II.2.28, VII.2.B, VIII.2, and VIII.3).	Prior to operating above 2546 MWt (98.4% RP).
12. Determine EQ service life for excore detectors (Attachment 5, Sections III.2.A and V.1.C).	Prior to operating above 2546 MWt (98.4% RP).
13. The ex-core neutron detectors are scheduled to be replaced (Attachment 5, Section V.I.C).	Unit 1: Fall 2010 Refueling Outage. Unit 2: Spring 2011 Refueling Outage.
14. Revise EOP setpoints (Attachment 5, Section VII.2.A).	Prior to operating above 2546 MWt (98.4% RP).
15. The UFM feedwater flow and temperature data will be compared to the feedwater flow venturis output and the feedwater RTD output (Attachment 5, Section I.1.D.2.1).	Prior to operating above 2546 MWt (98.4% RP).