VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 29, 2010

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

Serial No.	10-266
NLOS/GDM	R2'
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
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated January 27, 2010 (Serial No. 09-223), Dominion submitted a measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) for Surry Power Station Units 1 and 2 to increase the rated power of each unit by approximately 1.6%. On the same date, proprietary information [for the Cameron ultrasonic flowmeter (UFM)] required to support the LAR was submitted under separate cover (Serial No. 09-223A). By letter dated February 4, 2010 (Serial No. 09-223B), Dominion provided additional supporting information to facilitate the NRC's review of the plant accident analyses updates required by and discussed in the MUR power uprate LAR.

In a letter dated April 14, 2010, the NRC provided a request for additional information (RAI) associated with the Dominion MUR power uprate LAR submittal. Dominion's response to the NRC's RAI is contained in Attachment 1. As a result of the response to the RAI questions, Dominion has identified an additional commitment to those included in the list of regulatory commitments contained in Attachment 6 of the Surry MUR power uprate LAR dated January 27, 2010. The additional regulatory commitment is provided in Attachment 2. It should also be noted that the completion date for Commitment No. 9, *UFM commissioning and calibration will be completed*, in Attachment 6 of the Surry MUR power uprate LAR has been changed from "April 2010" to "Prior to operating above 2546 MWt (98.4% RP)" to facilitate integration of this activity into the overall MUR power uprate license amendment implementation effort.

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

Sincerely,

J. Alan Price
Vice President - Nuclear Engineering

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

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he 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 his knowledge and belief.

Acknowledged before me this 29TH day of April, 2010.

My Commission Expires: MAY 31, 2010.

Vicki L. Hull
Notary Public

Attachments:

1. Response to NRC Request for Additional Information
2. Regulatory Commitment Associated with the Response to the NRC Request for Additional Information

cc: U.S. Nuclear Regulatory Commission
Region II
245 Peachtree Center Avenue, NE Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
Surry Power Station

State Health Commissioner
Virginia Department of Health
James Madison Building – 7th Floor
109 Governor Street
Suite 730
Richmond, Virginia 23219

Ms. K. R. Cotton
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Dr. V. Sreenivas
NRC Project Manager
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 08 G-9A
11555 Rockville Pike
Rockville, Maryland 20852-2738

Attachment 1

<u>RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION</u>

<u>MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE LICENSE AMENDMENT REQUEST</u>

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

Response to Request for Additional Information
Measurement Uncertainty Recapture Power Uprate License Amendment Request

Surry Power Station Units 1 and 2

By letter dated January 27, 2010 (Serial No. 09-223), Virginia Electric and Power Company (Dominion) submitted a measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) for Surry Power Station (Surry) Units 1 and 2 to increase the rated power of each unit by approximately 1.6%. In a letter dated April 14, 2010, the NRC provided a request for additional information associated with the Dominion MUR power uprate LAR submittal. Dominion's response to the NRC's request is provided below.

Vessels and Internals Integrity Branch

1. *Attachment 5, Section IV, "Mechanical/Structural/Material Component Integrity and Design," requires additional information. Table Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides the NRC staff's basis for evaluating the potential for extended power uprates to induce aging effects on reactor vessel (RV) internals. Depending on the magnitude of the projected RV internals fluence, Table Matrix-1 may be applicable to the MUR application. In the "Notes" to Table Matrix-1, the NRC staff states that guidance on the neutron irradiation-related threshold for irradiation-assisted stress corrosion cracking (SCC) for pressurized water reactor (PWR) RV internal components are given in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," and WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals." [T]he "Notes" to Table Matrix-1 state that for thermal and neutron embrittlement of cast austenitic stainless steel, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs. Discuss your management of the above-mentioned aging effects on RV internals in light of the guidance in BAW-2248A and WCAP-14577, Revision 1-A. Please also confirm whether you have established an inspection plan to manage the age-related degradation in the Surry Units 1 and 2 RV internals, or whether you have participated in the industry's initiatives on age-related degradation of PWR RV internals.*

   <u>Dominion Response</u>

   Dominion confirms that it participates in industry initiatives on age-related degradation of PWR RV internals. Involvement has included participation on the EPRI Materials Reliability Program (MRP) committees that have researched the relevant degradation mechanisms, determined their importance to the functioning of each of the assemblies of the reactor internals, and produced the aging management recommendations contained in MRP-227 Revision 0. In support of NRC acceptance of this document, Dominion is presently participating in the small group of MRP members that are responding to questions from NRC reviewers. In Dominion letter No. 01-282, entitled "Virginia Electric and Power Company Surry and North Anna Power Stations Units 1 and 2 License Renewal Applications – Submittal" dated May 29, 2001, Dominion stated that inspections will be performed to implement industry recommendations. Inspection plans are currently under development.

BAW-2248A is not applicable to Surry Units 1 and 2. WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," was used by Dominion for developing the Surry License Renewal Application. Within the Surry license renewal application, Table 3.1.3-W1, "WCAP-14577, Rev. 1-A, FSER Response to Applicant Action Items", confirms the applicability of WCAP-14577, Rev 1-A to Surry Units 1 and 2.

In the intervening years since issuance of WCAP-14577, further materials research and functional requirements review carried out under EPRI sponsorship have resulted in additional recommendations for aging management contained in MRP-227 Revision 0. These recommendations are specifically configured for the combinations of degradation mechanisms and component functional requirements of the reactor internals. The MRP document addresses the aging mechanisms that are listed in WCAP-14577, including thermal and neutron embrittlement of cast austenitic stainless steel, SCC, and void swelling. It also has recommendations specific to plant design features. Therefore, in accordance with the commitment to consider industry guidance, the "focused inspection" planned for Surry prior to the period of extended operation will be consistent with the augmented examinations required by MRP-227 Revision 0. In addition, the ASME Section XI examination program for internals will continue to apply to other components not requiring augmented examinations.

NRC RS-001 Revision 0, "Review Standard for Extended Power Uprates", is focused on large power increases of up to 20%, as compared to the 1.6% MUR power uprate requested for Surry Units 1 and 2. In contrast, the Surry MUR results in very small changes to aging parameters such as temperature and neutron flux, and not at a level requiring revised aging management activities. Given the short period of time between implementation of the uprate and the beginning of the period of extended operation for each Surry reactor, the aging management program for the reactor internals as currently committed for Surry Units 1 and 2 is adequate. Therefore, RS-001 does not apply to the Surry MUR.

Instrumentation and Controls Branch

1. *License Amendment Request (LAR) Attachment 5, Page 20, Section I.1.G ("Completion Time and Technical Basis"), the LAR cites recent inspection of the feedwater flow venturis at North Anna as evidence that venturi fouling is unlikely to occur during any 48 hour period (during which the flow venturi may be used if the*

unified fracture mechanics [sic] [ultrasonic flowmeter] UFM is declared non-functional). With the understanding that both plants are of similar vintage from the same vendor, please provide information regarding the similarities between the feedwater flow venturis installed in the North Anna and Surry units. Additionally, have the Surry unit feedwater flow venturi been similarly inspected? If so, how recent was the inspection and were there any observations made regarding fouling?

Dominion Response

Surry has three feedwater venturis on each unit (1-FW-FE-1476, 1-FW-FE-1486, 1-FW-FE-1496, 2-FW-FE-2476, 2-FW-FE-2486 and 2-FW-FE-2496). The feedwater flow venturis at North Anna Power Station and Surry were manufactured by the same vendor but with different sizes and model numbers. The Surry feedwater venturis were replaced by Design Change 94-055 (Unit 1) in October 1995 and Design Change 90-034 (Unit 2) in November 1991. Periodic inspections and cleanings have been performed on each of the venturis since that time in accordance with the preventive maintenance program. The inspection and cleaning is performed by procedure 0-MPM-1010-01, "Feedwater Venturi Inspection," and consists of a check of the interior of the venturi, the high and low pressure taps, and an inspection for damage, corrosion, erosion or other abnormalities. This check is performed by Maintenance and System Engineering personnel. The internals are then cleaned. A quality inspection hold point is performed prior to system close out to check for foreign material. The cleaning procedure includes a caution to preserve the venturi integrity that reads "Venturi interior finish can be damaged from wire brush by scratching. Care must be exercised during venturi cleaning to prevent scratching venturi interior surface."

The following is a list of the years the venturis were inspected and the most recent preventive maintenance work order history and the results found for each venturi:

Feedwater Flow Element	Years Inspected
1-FW-FE-1476	1998, 2000, 2003, 2006
1-FW-FE-1486	1998, 2000, 2003, 2007
1-FW-FE-1496	1998, 2000, 2003, 2009
2-FW-FE-2476	1995, 1996, 1999, 2001, 2005, 2009
2-FW-FE-2486	1995, 1996, 1999, 2001, 2005, 2009
2-FW-FE-2496	1995, 1996, 1999, 2001, 2005

- 1-FW-FE-1476 (U1 A Steam Generator (S/G) flow element) was last inspected in May 2006. The as-found condition of the flow element was satisfactory. Slight pitting was noted at the 7 to 9 o'clock position, which was determined by Engineering to have no effect on venturi operation. This venturi is scheduled to be inspected during the 2010 refueling outage.

- 1-FW-FE-1486 (U1 B S/G flow element) was last inspected in October 2007. The as-found condition of the flow element was satisfactory. The internals were cleaned.

- 1-FW-FE-1496 (U1 C S/G flow element) was last inspected in April 2009. The as-found condition of the flow element was excellent. There was no evidence of corrosion, erosion or other damage. The internals were cleaned and closeout inspection was satisfactory.

- 2-FW-FE-2476 (U2 A S/G flow element) was last inspected in November 2009. The as-found condition of the flow element was satisfactory – no degradation was found. High and low pressure lines were verified to be clear.

- 2-FW-FE-2486 (U2 B S/G flow element) was last inspected in November 2009. The as-found condition of the flow element was satisfactory.

- 2-FW-FE-2496 (U2 C S/G flow element) was last inspected in April 2005. The as-found condition of the flow element was satisfactory. This flow element had a higher buildup of magnetite (carbon-like coating on inside of pipe) than the other two flow elements for Unit 2 when inspected in 2005. The high side sensing line port also had a buildup of magnetite. The coating was removed by cleaning. No other discrepancies were noted. The next inspection of 2-FW-FE-2496 is scheduled for the 2011 refueling outage.

In addition to the periodic venturi inspections, Engineering performs monthly trending of the feedwater flow for the three loops. There have been no adverse trends in the monthly flow data between inspections of the flow elements. Engineering also performs a comparison of the reactor power calorimetric to alternate power indications at 96% rated power during power escalation (e.g., after each refueling outage). This evaluation includes a review of the feedwater flow calorimetric and steam flow calorimetric indicators. The difference between the main feedwater and the main steam power calorimetric has been stable, indicating no fouling or defouling of the feedwater venturis on Surry Units 1 and 2.

2. *LAR Attachment 5, Page 20, Section I.1.G ("Completion Time and Technical Basis"), the LAR states that a feedwater flow transmitter drift study was used as a basis for determining that transmitter drift over any 48-hour period would be negligible (during which the flow venturi may be used if the UFM is declared non-functional). Please provide a reference for the cited study.*

<u>Dominion Response</u>

The feedwater flow transmitter drift data was obtained from Surry Instrument Periodic Test (IPT) procedures (1-IPT-CC-FW-F-476, -477, -486, -487, -496 and

-497) that document the as-found and as-left transmitter output voltage during each refueling outage. The data from the six feedwater flow transmitters was reviewed for two consecutive 18-month operating cycles for Surry Unit 1 spanning from April 2006 to April 2009. The as-left and as-found output voltages at the 100% power differential pressure point were converted to mass flow rates, and the percent change in main feedwater flow was calculated for each transmitter over the cycle. The cycle-average change in main feedwater flow was 0.017% and 0.014% for the first and second cycles, respectively, and the maximum individual transmitter output change was 0.05% flow. The transmitter drift analyses are documented in an internal engineering document (Engineering Transmittal ET-NAF-09-0013) that conforms to 10 CFR 50 Appendix B quality assurance requirements. The completed IPTs from April 2006, October 2007, and April 2009 are stored in the Surry records system. Also, the applicable transmitter data sheets are attached to the Engineering Transmittal. The evaluation using Unit 1 data was determined to be applicable for Unit 2 (both units have the same transmitter types). It was concluded that the feedwater flow transmitter drift over any 48-hour period would be negligible for Surry Units 1 and 2.

3. *LAR Attachment 5, Page 22, Section I.1.H ("Actions for Exceeding Completion Time and Technical Basis"), the LAR specifically notes that the Surry units have the option to use either steam or feed flow as input to the calorimetric calculation when the UFM is non-functional. The LAR also notes that "within the first 48 hours after the identification of a non-functional UFM, normalized main feed flow will be used." Section 3.3.5 of the Technical Requirements Manual indicates that in the first 48 hours after the UFM is discovered non-functional, the normalized feedwater venturi system would be used. Is there an intention, as part of this LAR, to be able to use the main steam flow venturi for calorimetric calculations during the first 48 hours following UFM non-functionality to maintain power above 2,546 MWt? If so, is the steam flow venturi measurement calibrated to the UFM?*

Dominion Response

The main steam venturi-based calorimetric will not be used for power calorimetric calculations to support plant operation above 2546 MWt. Technical Requirements Manual (TRM) 3.3.5 Required Action A.1 (provided in Attachment 4 of the Surry MUR power uprate LAR dated January 27, 2010) allows use of only the Normalized Feedwater Venturi System during the 48 hour completion time with a non-functional UFM. The steam flow-based calorimetric may be used for operation ≤ 2546 MWt consistent with current plant procedures. TRM 3.3.5 Required Action B.2 directs the use of the Feed (not normalized) or Steam Venturi System after core power is reduced ≤ 2546 MWt after the 48 hour completion time has passed.

Reactor Systems Branch

1. *Departure from Nucleate Boiling Ratio (DNBR) Analyses*

The licensee evaluated the majority of these transients for the effect of the increased power level on DNBR. The evaluation included scaling the transient DNBR response by core power level and allocating a DNBR margin based on the characterization of the power uprate in terms of fractional effect on DNBR, as determined by the power evaluation. The evaluation did not consider other DNBR-significant parameters that could change as a result of the requested uprate, including rod surface heat flux, core/channel inlet enthalpy, core flow rate, and reactor coolant system temperature.

(a) Please explain the effect of the following parameters on the DNBR, and discuss how the DNBR margin evaluation accounted for each: (1) fuel rod surface heat flux; (2) core and channel inlet enthalpy; (3) core flow rate; and (4) reactor coolant system temperature.

Dominion Response

The MUR power uprate evaluation considered the effects of the identified plant parameters on the calculation of departure from nucleate boiling ratio (DNBR). The summary of our evaluation is provided.

(1) An increase in the nominal fuel rod surface heat flux decreases the DNBR. The effect on the fuel rod surface heat flux increase is explicitly accounted for in the COBRA statepoint analysis. The fuel rod surface heat flux is directly proportional to total core power, and a 1.7% increase in core power produces a 1.7% increase in fuel rod surface heat flux in the COBRA analysis model. The 1.7% increase in heat flux produces the 3.3% decrease in DNBR. A 1.7% core power and fuel rod surface heat flux increase was selected to bound the approximately 1.61% MUR power uprate from 2546 MWt to 2587 MWt.

(2) An increase in the core inlet temperature decreases the DNBR. Core and channel inlet enthalpy were not changed in the DNBR analysis for the MUR uprate. This is conservative for the MUR uprate with constant reactor coolant system (RCS) average temperature (Tavg) of 573°F. As core power increases with constant Tavg, core and channel inlet temperature (enthalpy) decrease and provide an increase in DNBR. This DNBR benefit was ignored for conservatism in the MUR power uprate assessment.

(3) An increase in the core mass flow rate increases the DNBR. Core mass flow rate was not changed in the DNBR analysis for the MUR uprate. The full-power statistical DNBR analyses assume the RCS is at the minimum measured volumetric flow (MMF) rate. The reduction in core inlet temperature discussed above in part (2) at a constant MMF would increase the fluid density, increase

the core mass flow rate, and increase DNBR. This DNBR benefit was ignored for conservatism in our MUR power uprate assessment.

(4) The current full-power RCS Tavg is 573°F, and Surry plans to maintain the same Tavg at the MUR power level of 2587 MWt. As discussed above in parts (2) and (3), the MUR power uprate from 2546 MWt to 2587 MWt core power with a constant RCS Tavg of 573°F will reduce core inlet enthalpy and increase the vessel mass flow. The DNBR benefits from these changes were ignored in determining the 3.3% DNBR penalty.

Core power is the only plant parameter in the Updated Final Safety Analysis Report (UFSAR) Chapter 14 hot full power statistical DNBR analyses not bounded at MUR power uprate conditions. The DNBR margin evaluation conservatively assumes no benefit attributed to the core and channel inlet enthalpy, core mass flow rate, and RCS temperature at MUR uprate conditions. The experience described below has shown that a full reanalysis (transient response and DNBR) at a 1.7% MUR power uprate produces a smaller DNBR increase than the 3.3% statepoint penalty that was developed for Surry. The reanalysis of the rod withdrawal at power (RWAP) event from UFSAR Section 14.2.2 was performed to demonstrate adequate reactor protection with the proposed change to the K_3 pressure constant in the Overtemperature ΔT (OTΔT) reactor trip. The reanalysis is discussed in Section VIII of Attachment 5 of the LAR dated January 27, 2010 [Reference 1a]. For a 1.7% increase in core power from 2546 MWt to 2589.3 MWt, the minimum DNBR decreased by 2.4% from 1.68 to 1.64 (above the DNBR limit of 1.46). This analysis explicitly accounted for the decrease in core inlet temperature and increase in core mass flow rate corresponding to RCS Tavg of 573°F at 2589.3 MWt core power (versus 2546 MWt in the analysis of record). With the exception of the RWAP event which is specifically reanalyzed, the 3.3% DNBR penalty applied against full-power statistical DNBR events is conservative for the range of thermal-hydraulic conditions expected for accidents initiated from the MUR uprate power level of 2587 MWt (1.61% increase above 2546 MWt).

(b) Provide a detailed DNBR margin evaluation to substantiate the claim that there is adequate retained DNBR margin to account for the effect of the requested power uprate.

<u>Dominion Response</u>

Thermal-hydraulic evaluations were reviewed for the three most recent cycles for each Surry unit (six total cycles). Each calculation documents the DNBR penalties applicable to the core reload, identifies the DNBR limits and retained DNBR margin, and demonstrates positive retained DNBR margin for each Surry UFSAR Chapter 14 DNB event. This evaluation of cycle-specific DNBR margins is performed consistent with Dominion's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A [Reference 1b]. DNBR penalties against retained margin are classified as 1) generic fuel design issues (e.g., fuel rod bow), 2) cycle-specific violations of limits (e.g., unbounded power shapes or peaking factors), or 3) plant operating conditions. The use of retained DNBR margin is consistent with Dominion's NRC-approved Statistical DNBR Evaluation Methodology in VEP-NE-2-A [Reference 1c] and is discussed in Surry UFSAR Section 3.4.3.5 [Reference 1d]. The Reference 1b and 1c topical reports are identified as References 1 and 3a in Surry Technical Specification 6.2.C, "Core Operating Limits Report" [Reference 1e]. Six cycles of design data provide an adequate overview of trends in unbounded cycle-specific parameters to assess available DNBR margin for future MUR uprate cycles.

Table 1 summarizes the DNBR penalties that are expected during MUR uprate cycles. The 13.0% retained DNBR margin is applicable to statistical DNB analyses of Surry Improved Fuel using the WRB-1 correlation (difference between safety analysis limit of 1.46 and statistical design limit of 1.27), as discussed in Section II.2 of Attachment 5 of the Surry MUR power uprate LAR dated January 27, 2010 [Reference 1a]. A DNBR margin summary is provided for two groups of UFSAR Chapter 14 events: 1) accidents that are protected by the OTΔT reactor trip; 2) accidents that credit other reactor protection (non-OTΔT events). These events are tracked separately, because the cycle-specific power shape penalties are different. Table 1 shows adequate retained DNBR margin is available to accommodate the 3.3% penalty for a bounding 1.7% power uprate.

Table 1: Typical Retained DNBR Margins for Full Power Statistical DNB Events

	OTΔT Events	**Non-OTΔT Events**
Retained DNBR Margin	13.0%	13.0%
Generic Fuel Design Issues		
Fuel Rod Bow	2.8%	2.8%
Cycle-Specific Unbounded Parameters		
Power Shapes*	0.8%	4.5%
Plant Operating Conditions		
Bypass Flow	1.7%	1.7%
1.7% Power Uprate	3.3%	3.3%
Margin = Retained Margin – Penalties	+ 4.4 %	+ 0.7 %

* The maximum power shape penalty from six recent Surry cycles is reported. Future cycles are expected to produce smaller penalties due to a change in the core loading pattern strategy that reduces the magnitude of unbounded cycle-specific power shapes.

References

1a. Letter from L. N. Hartz (Dominion) to USNRC, *Virginia Electric and Power Company (Dominion), Surry Power Station Units 1 and 2, License Amendment Request, Measurement Uncertainty Recapture Power Uprate*, Serial No. 09-

223, Rev. 1, January 27, 2010. [NRC ADAMS Accession Number ML100320264]

1b. Topical Report VEP-FRD-42, Revision 2.1-A, *Reload Nuclear Design Methodology*, August 2003.

1c. Topical Report VEP-NE-2-A, *Statistical DNBR Evaluation Methodology*, June 1987.

1d. Surry Updated Final Safety Analysis Report, Revision 41.

1e. Surry Power Station Units 1 and 2 Technical Specifications.

2. *<u>Items Within the Reload Licensing Methodology Scope</u>*

For the Excessive Heat Removal due to Feedwater System Malfunctions, the Excessive Load Increase, the Loss of Reactor Coolant Flow, and the Loss of External Electrical Load transients, provide either explicit analyses, or the following information outlined in RIS 2002-03, Attachment 1, Section III:

(a) Identify the accident/transient 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 uprate power level, if NRC review is deemed necessary by the criteria in Title 10 of the Code of Federal Regulations *(10 CFR) 50.59; and*

(d) Provide a reference to the NRC's approval of the plant's reload methodology.

<u>Dominion Response</u>

For the applicable UFSAR Chapter 14 events, Surry will re-analyze the transient consistent with Dominion's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A [Reference 2a] prior to implementation of the MUR power uprate. This topical report is identified as Reference 1 in Surry Technical Specification 6.2.C, "Core Operating Limits Report" [Reference 2b]. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A [Reference 2c]:

- Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only);

- Section 14.2.8 - Excessive Load Increase Incident;
- Section 14.2.9 - Loss of Reactor Coolant Flow; and
- Section 14.2.10 - Loss of External Electrical Load

The Surry UFSAR description for each Chapter 14 DNBR analysis will be modified to reflect the results of the MUR uprate analysis. The UFSAR updates will be performed in accordance with Commitment #8 in Attachment 6 of the Surry MUR power uprate LAR dated January 27, 2010 [Reference 2d].

<u>References</u>

2a. Topical Report VEP-FRD-42, Revision 2.1-A, "Reload Nuclear Design Methodology," August 2003.

2b. Surry Power Station Units 1 and 2 Technical Specifications.

2c. Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

2d. Letter from L. N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), Surry Power Station Units 1 and 2, License Amendment Request, Measurement Uncertainty Recapture Power Uprate," Serial No. 09-223, Rev. 1, January 27, 2010. [NRC ADAMS Accession Number ML100320264]

3. <u>*Steam Line Break*</u>

Evaluate the effects of the requested uprate against a hot full power main steam line break (MSLB) and demonstrate that the transient remains non-limiting.

<u>Dominion Response</u>

Section 14.3.2 of the Surry UFSAR [Reference 3a] describes the basis for why a main steam line break (MSLB) at power is bounded by the DNBR analysis at zero power. Following a trip at power, the reactor coolant system (RCS) contains more stored energy than at no load, the average coolant temperature is higher than at no load, and there is appreciable stored energy in the fuel. The additional stored energy is removed via the cooldown caused by the MSLB before no-load conditions are reached. After the additional stored energy above no load is removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that is initiated from hot zero power conditions. Since the initial SG liquid mass is significantly larger at no load than at full power, the magnitude and duration of the RCS cooldown are less for a MSLB at power. For the MUR uprate, the increase in core thermal power would increase the core stored energy by a small amount, RCS average coolant temperature will be the same, and the steam generator pressure and liquid mass will not change significantly. Collectively, these small effects would

produce essentially the same MSLB response at either 2546 MWt or 102% of 2546 MWt core power.

The Surry UFSAR "at power" discussion is supported by generic MSLB analyses performed by Westinghouse in WCAP-9226 [Reference 3b]. Analyses were performed for a 3-loop PWR similar to Surry at 0%, 30%, 70%, and 102% of 2785 MWt core power. The maximum analyzed power level of 102% of 2785 MWt is bounding for Surry's current rated power of 2546 MWt and the MUR rated power of 2587 MWt. WCAP-9226 demonstrates that, for a MSLB at core power levels greater than Surry's, the reactor protection system provides adequate protection to ensure the DNB design basis is not violated prior to and immediately following a reactor trip. Furthermore, WCAP-9226 concludes that the limiting MSLB transient occurs from a zero power initial condition. The Surry UFSAR analysis basis with zero power as the limiting MSLB case was verified for the Surry core power uprate from 2441 MWt to 2546 MWt (Section 3.4.1 in Attachment 3 of Reference 3c) and remains valid for the MUR power uprate to 2587 MWt.

Even though the MSLB is classified as an ANS Condition IV event, Surry continues to meet the more stringent ANS Condition II DNBR limits for this event. This conclusion is reconfirmed for each reload in accordance with Section 3.3.4.4 of topical report VEP-FRD-42, Rev. 2.1-A [Reference 3d].

<u>References</u>

3a. Surry Updated Final Safety Analysis Report, Revision 41.

3b. WCAP-9226, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," Westinghouse Electric Corporation, January 1978.

3c. Letter from James P. O'Hanlon (Virginia Power) to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specification Changes to Accommodate Core Uprating," Serial No. 94-509, August 30, 1994.

3d. VEP-FRD-42, Revision 2.1-A, "Reload Nuclear Design Methodology," August 2003.

4. *<u>Control Rod Assembly Drop/Misalignment</u>*

For the Control Rod Assembly Drop/Misalignment transient, clarify whether cycle-specific confirmation of the dropped rod limit lines will consider uprated operation, and whether the confirmation is performed in accordance with NRC-approved reload licensing methodology.

If the confirmation is not performed in accordance with NRC-approved reload licensing methodology, provide a disposition for the Control Rod Assembly Drop/Misalignment transient that adheres to the guidance in Section III.3 of Attachment 1 to RIS 2002-03.

Dominion Response

The Surry cycle-specific confirmation of the dropped rod limit lines will consider the MUR uprated core power of 2587 MWt. Because the dropped rod limit lines are based on a core power of 2546 MWt with a DNBR safety analysis limit of 1.46, the 3.3% DNBR penalty (for a bounding 1.7% power uprate) will be applied on a reload basis for the COBRA analysis of the Surry Improved Fuel for the dropped rod event. The 3.3% DNBR penalty is conservative for the thermal-hydraulic conditions expected during the dropped rod event initiated from the MUR power level of 2587 MWt (1.61% increase above 2546 MWt). The cycle-specific confirmation of the Surry dropped rod limit lines is performed according to Section 3.3.4.2 in the NRC-approved topical report VEP-FRD-42, Revision 2.1-A [Reference 4a], which is identified as Reference 1 in Surry Technical Specification 6.2.C, "Core Operating Limits Report" [Reference 4b].

References

4a. VEP-FRD-42, Revision 2.1-A, "Reload Nuclear Design Methodology," August 2003.

4b. Surry Power Station Units 1 and 2 Technical Specifications.

5. *Licensing Basis Control*

10 CFR 50.71(e) promulgates requirements for updating the final safety analysis report (FSAR), stating, in part, that FSAR update submittals "shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the licensee pursuant to Commission requirement since the submittal of the original FSAR, or as appropriate, the last update to the FSAR under this section. The submittal shall include... all safety analyses and evaluations performed by the licensee either in support of approved license amendments...." In light of the statement, in selected notes to Table II-2, that "The UFSAR analyses of record for DNBR do not need to be updated," explain how adherence to 10 CFR 50.71 will be maintained following implementation of the requested MUR uprate.

Dominion Response

The statement in Table II-2 was intended to differentiate between the approach of using retained DNBR margin in lieu of revising or updating the explicit DNBR transient analysis that is presented in the UFSAR [Reference 5a]. The Surry UFSAR will be updated to identify the 3.3% DNBR penalty that is applied to each full-power statistical DNBR event that was analyzed at 2546 MWt core power. The penalty is applicable to COBRA analyses of the Surry Improved Fuel product. The following UFSAR sections will be modified:

- Section 14.2.4, Control Rod Assembly Drop/Misalignment
- Section 14.2.7, Excessive Heat Removal Due to Feedwater System Malfunction (Full Power Feedwater Temperature Reduction)
- Section 14.2.8, Excessive Load Increase Incident
- Section 14.2.9, Loss of Reactor Coolant Flow
- Section 14.2.10, Loss of External Electrical Load

As discussed in Section VIII of Attachment 5 of the Surry MUR power uprate LAR dated January 27, 2010 [Reference 5b], the Rod Withdrawal at Power event in UFSAR Section 14.2.2 was reanalyzed at a bounding core power of 2589.3 MWt (1.7% above 2546 MWt) and the results of the re-analysis will be added to UFSAR Section 14.2.2. The UFSAR updates described above will be performed in accordance with Commitment #8 in Attachment 6 of the LAR dated January 27, 2010 [Reference 5b].

<u>References</u>

5a. Surry Updated Final Safety Analysis Report, Revision 41.

5b. Letter from L. N. Hartz (Dominion) to USNRC, "Virginia Electric and Power Company (Dominion), Surry Power Station Units 1 and 2, License Amendment Request, Measurement Uncertainty Recapture Power Uprate," Serial No. 09-223, Rev. 1, January 27, 2010. [NRC ADAMS Accession Number ML100320264]

6. *<u>Transducer Replacement</u>*

In the submittal, the licensee states that they will generate transducer replacement procedures. It is unclear when the procedures will be finalized, whether before, or after implementation of an MUR.

<u>Dominion Response</u>

The transducer replacement procedure (EFP-18) is provided in both the Installation and Commissioning Manual and the Maintenance and Troubleshooting Manual. Dominion will incorporate this vendor procedure into Surry station procedures prior to operating above 2546 MWt (98.4% RP). [See Surry MUR LAR, Serial No. 09-223, Attachment 6, Commitment 3, January 27, 2010.]

7. *Software*

Describe the system software verification and validation program. How does the program ensure data from the UFM is appropriately analyzed and applied?

Dominion Response

The Leading Edge Flowmeter (LEFM) Software Verification and Validation (V&V) Program is designed to document the specification, implementation and testing of the LEFM software so that the software responds and performs in an expected and defined manner. The V&V Program is designed to meet EPRI TR-103291S-V1-3. The program defines the requirements, guides the implementation, and confirms proper implementation by test of the LEFM application software. The site specific software inputs (INI files) are measured and determined by the commissioning procedure (EFP-61) and the procedure governing the creation and documentation of the INI file (EFP-302-2). These procedures are performed under Cameron's QA program so that the calculations are independently reviewed and the INI file is entered into the revision control plan. The data is checked for consistency with laboratory calibration testing and documented in the System Uncertainty Calculation.

8. *Self-Verification Feature*

Explain the self-verification feature of the software.

Dominion Response

The LEFM self-verification features provide a comprehensive check of signal quality, timing, path lengths, speed of sound, velocity profiles, and measurement statistics. The values are compared and traceable through the System Uncertainty Analysis to ensure that the system remains within its design basis uncertainty. These features are described in detail in the Topical Reports ER-80P and ER-157P and in the LEFM Specification LEFM√103 contained in the Verification and Validation Program Documentation.

9. *Definition*

In the submittal, the licensee states that the software "continuously adjusts venturi flow coefficients and feedwater resistance temperature detector (RTD) temperatures." Define the term continuously.

Dominion Response

The term continuously is defined as once per minute based on rolling, one hour averages. The software continuously calculates feedwater (FW) flow normalization factors and feedwater temperature and pressure biases to calculate Filtered Normalized FW Flow.

10. *Preventive Maintenance Program*

The licensee states that they will develop a preventive maintenance program. When is the program scheduled to be developed?

Dominion Response

Guidelines and recommendations for Maintenance are contained in the LEFM Maintenance and Troubleshooting Manual. Dominion will incorporate these guidelines and recommendations into Surry station procedures prior to operating above 2546 MWt (98.4% RP). [See Surry MUR LAR, Serial No. 09-223, Attachment 6, Commitment 3, January 27, 2010].

11. *Calibration and Maintenance*

Are calibration and maintenance procedures established? If not, when will the procedures be finalized?

Dominion Response

The affected calibration and maintenance procedures are being revised as part of the MUR implementation design change packages (DCPs) (DCPs SU-08-0027 and SU-08-0028). These procedures are scheduled to be finalized prior to operating above 2546 MWt (98.4% RP). [See Surry MUR LAR, Serial No. 09-223, Attachment 6, Commitment 3, January 27, 2010].

12. *Conditions Adverse to Quality*

Define "adverse to quality" with respect to reporting deficiencies to the manufacturer and what actions are implemented if a condition "adverse to quality" is found to exist.

Dominion Response

Section 4, Terms and Definitions, of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications," defines "condition adverse to quality" as "an all-inclusive term used in reference to any of the following: failures, malfunctions, deficiencies, defective items, and nonconformances. A significant condition adverse to quality is one which, if uncorrected, could have a serious effect on safety or operability." The Dominion Quality Assurance Topical Report DOM-QA-1 entitled, "Nuclear Facility Quality Assurance Program Description," states, "The Company has established and implements corrective action programs, procedures, and processes to assure that conditions adverse to quality at Company nuclear facilities are promptly identified and corrected." Upon discovery of a deficiency or condition adverse to quality, Dominion procedures require entry of the condition into the Central Reporting System, determination of the cause of the condition, and performance of appropriate corrective actions. These actions may include, but are not limited to, notification in accordance with 10 CFR 21, *Reporting of Defects and Noncompliance*.

13. Power Calorimetric

Please explain the differences when using steam flow in the power calorimetric rather than feed flow.

Dominion Response

The constituents of the current total calorimetric uncertainty are the uncertainties for the flow venturi discharge coefficient, venturi differential pressure measurement, feedwater temperature measurement, moisture carryover, and steam pressure measurement. The current power calorimetric uncertainties at Surry Units 1 and 2 are 0.90% rated power using main feedwater flow and 1.21% rated power using main steam flow. These calculations are based on input from the plant computer system. Both methods provide a calorimetric uncertainty that is less than the 2% used in the deterministic safety analyses. As stated in the response to Instrumentation and Controls Branch Question #3, the main steam venturi-based calorimetric will not be used to support plant operation above 2546 MWt.

Accident Dose Branch

1. *Section III.2.B.1.3 of Attachment 5 to the January 27, 2010, Surry MUR power uprate LAR (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100320264) provides information about the atmospheric dispersion factors (X/Q values) used in the steam generator tube rupture analysis. This section states that the control room (CR) and low population zone (LPZ) X/Q values remain unchanged from the current licensing basis and cites Surry Amendment No. 230 (ADAMS Accession No. ML020710159) dated March 8, 2002, as the reference. Table III.5 of Attachment 5 lists the CR X/Q values, which were also used for the MSLB accident dose assessment.*

NRC staff agrees that the LPZ X/Q values listed in Table III-5 of the LAR are those in the safety evaluation (SE) that supports Amendment No. 230, but was unable to find the CR X/Q values in the SE. Therefore, please provide a reference for and discussion of the CR X/Q values, including confirmation that use of the CR X/Q values for the MUR power uprate LAR remains unchanged from their use in the current licensing basis analysis.

Dominion Response

Time dependent X/Q values for the Main Steam Line Break (MSLB) and Steam Generator Tube Rupture (SGTR) control room emergency intake were originally docketed as part of the Control Room Dose Calculations/Habitability Assessment (Reference 1c). The MSLB and SGTR control room X/Q values used in the current design basis were docketed as part of the stretch power uprating (References 1a and 1b). A discussion of these values can be found in Sections 3.7.2.2.3 (MSLB) and 3.7.2.3.3 (SGTR) of Reference 1a. The MSLB and SGTR control room X/Q values are based on the Murphy and Campe methodology.

The emergency control room intake X/Q value (3.79E-03 sec/m^3) is a 0-8 hour X/Q value that was applied over the duration of the MSLB release and for the duration of the SGTR after the control room is isolated. The emergency control room intake X/Q value is presented in Table 3.7.2.2-4 of Reference 1a. The normal control room intake X/Q value (7.71E-03 sec/ m^3) is a 0-8 hour X/Q value that was applied to the SGTR prior to isolation by an SI signal, which occurs at 0.0687 hours (247 seconds). The normal control room intake X/Q value is presented in Section 3.7.2.3.3 of Reference 1a.

The control room X/Q values remain unchanged from the current license basis analysis.

References

1a. 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. (Reference III-2 in Attachment 5 of the Surry MUR LAR, Serial No. 09-223, January 27, 2010.)

1b. 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 90365)," Serial No. 95-405, August 3, 1995. (Reference III-3 in Attachment 5 of the Surry MUR LAR, Serial No. 09-223, January 27, 2010.)

1c. 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. (Reference III-7 in Attachment 5 of the Surry MUR LAR, Serial No. 09-223, January 27, 2010.)

<u>Mechanical and Civil Engineering Branch</u>

1. *Section IV.1.A.iv in Attachment 5 of Reference 1 [of the LAR] states that operation at the proposed MUR conditions will have an insignificant impact on the analyses and evaluations for the reactor coolant loop piping, primary equipment nozzles, primary equipment supports, Class 1 auxiliary piping lines attached to the reactor loop piping, and the Class 1 auxiliary line branch nozzles attached to the reactor loop piping. However, the LAR request does not indicate whether these piping system components and supports are still bounded by the existing design basis analyses. Please verify whether the current analyses of record (AOR) remain bounding for the aforementioned reactor coolant piping components and supports. If the AOR is not bounding, wholly or in-part, please provide the updated analyses results for the reactor coolant piping components and supports which are not bounded under the proposed MUR uprate conditions.*

   <u>Dominion Response</u>

   For the MUR power uprate conditions, the current analyses of record (AOR) remain bounding for the reactor coolant loop, primary equipment nozzles, primary equipment supports, auxiliary piping lines attached to the reactor loop, and the auxiliary branch nozzles attached to the reactor loop piping. As identified in the LAR submittal (Serial No. 09-223) dated January 27, 2010, RCS piping is designed to USAS B31.1 code requirements.

2. *Section IV.1.A.v in Attachment 5 of Reference 1 [of the LAR] indicates that the Balance-of-Plant (BOP) piping systems were reviewed to determine what impact the proposed MUR uprate conditions would have on the abilities of the various BOP piping systems to continue operating at MUR power uprate levels. Accordingly, change factors based on thermal, pressure, and flow rate variances between the current and proposed MUR power uprate levels were used to determine whether the current AOR remains bounding for the BOP piping systems within the scope of the MUR power uprate LAR. Please address the following items regarding the BOP piping acceptability:*

(a) *Please clarify the following statement from page 96 of Attachment 5 of Reference 1 [of the LAR], "The changes are acceptable." Please indicate whether acceptability refers to all BOP piping system change factors remaining below 1.0; whether some systems were above 1.0, but were found acceptable based on an updated analysis for the system at the proposed MUR uprate conditions; or whether the current AOR remains bounding for all BOP piping systems considered within the scope of this LAR.*

Dominion Response

Changes in temperature and pressure, as a result of the proposed MUR power uprate, were reviewed and determined to be insignificant for the BOP piping systems except for the Main Steam and Steam Dump, Extraction Steam, Condensate, Feedwater, Heater Drain, and Service Water Systems. For these systems, the stresses were proportionally adjusted upward based on the change factors and compared with the allowable values. The adjusted stresses are within applicable allowable values.

An increase in mass flow rate will usually affect stresses resulting from fluid transient events such as pump trip, sudden valve closures, etc. A review was performed to identify the BOP piping affected by an increase in mass flow rate. It was determined that only the Main Steam System piping between the steam generators and the turbine trip stop valves (TSVs), which are considered fast acting valves and therefore capable of imposing significant fluid transient loadings in the system, would be affected. An analysis of this piping was performed to qualify the piping for this loading condition, since this loading condition had not been explicitly analyzed previously. The stresses were determined to be within the applicable allowable values.

Therefore, the changes in temperature, pressure and flow rate related to the MUR power uprate were determined to be acceptable.

(b) *In concert with the response to part (a) above, please indicate which, if any, systems had a change factor above 1.0 and indicate whether the thermal, pressure, or flow variance was the limiting parameter for these systems.*

Dominion Response

As identified in the response to part (a), the change factors for thermal and pressure variance were calculated for the Main Steam and Steam Dump, Extraction Steam, Condensate, Feedwater, Heater Drain and Service Water Systems. Change factors for thermal variance for portions of the Main Steam and Steam Dump, Extraction Steam, Condensate, Feedwater, Heater Drain and Service Water Systems were higher than 1.0. Change factors for pressure variance for portions of the Extraction Steam and Heater Drain Systems were also higher than 1.0. As noted in the response to part (a) above, only the Main Steam System piping required analysis for fluid transient loads, and the stresses due to fluid transient loads were determined to be within allowable values.

(c) *Based on the response to part (b) above, please summarize the results of the additional evaluations performed for the affected systems and indicate whether these systems remain bounded by the current AOR.*

Dominion Response

The maximum change factors for the thermal and pressure variance for the Main Steam and Steam Dump, Extraction Steam, Condensate, Feedwater, Heater Drain, and Service Water Systems are tabulated below for Surry Units 1 and 2.

	Unit 1		Unit 2	
	Temp	Press	Temp	Press
Main Steam and Steam Dump System	1.031	≤ 1	1.036	≤ 1
Extraction Steam System	1.053	1.06	1.053	1.053
Condensate System	1.204	≤ 1	1.191	≤ 1
Feedwater System	1.019	≤ 1	1.021	≤ 1
Heater Drain System	1.118	1.104	1.112	1.098
Service Water System	1.408	1	1.408	1

The maximum change factors for thermal variance for the Condensate, Heater Drain, and Service Water Systems were 1.204, 1.118 and 1.408, respectively. However, the portions of these systems with these maximum change factors are low temperature sub-systems (less than or equal to 125.9°F), and the range of temperature increase was approximately 10°F or less. The maximum change factors for thermal variance for the remaining systems were less than 1.07. These changes are not bounded by the current AOR, but were dispositioned as indicated in the response to part (d) below.

Change factors for pressure variance for the Extraction Steam and Heater Drain Systems were higher than 1.0. Specifically, the maximum change factors for pressure variance for portions of the Extraction Steam and Heater Drain Systems were 1.06 and 1.104, respectively. These changes are not bounded by the current AOR, but were dispositioned as indicated in the response to part (d) below.

(d) *Based on the response to part (c) above, please provide the updated analyses results for BOP piping systems whose current AOR is not bounding at the proposed MUR uprate conditions.*

Dominion Response

Since the change factors were higher than 1.0 for portions of the Main Steam and Steam Dump, Extraction Steam, Condensate, Feedwater, Heater Drain, and Service Water Systems, the stresses in the affected subsystems were proportionally adjusted upward based on the change factors. The prorated stresses were determined to be within allowable values.

As noted in our response to part (a) above, an analysis was performed of the affected Main Steam System piping to qualify the piping for the loading condition associated with an increase in mass flow rate. The stresses calculated in this analysis were determined to be within applicable allowable values.

Fire Protection Branch

1. *The NRC staff notes that Attachment 5 to the LAR, Section II.2.31, "Safe Shutdown Fire Analysis (Appendix R Report) UFSAR 9.1," states that "...*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 NRC staff requests the licensee to verify that (1) the MUR power uprate will not require any new operator actions; (2) any effects from additional heat in the plant environment from the increased power will not prevent required post fire operator manual actions, as identified in the Surry fire protection program, from being performed at and within their designated time; and (3) procedures and resources necessary for systems required to achieve and maintain safe-shutdown will not change and are adequate for the MUR power uprate.*

   Dominion Response

   (1) Section VII.1 in Attachment 5 of the Surry MUR power uprate LAR dated January 27, 2010 summarizes the review of the operator actions assumed in the safety analyses, including the Appendix R fire safe shutdown analyses. The Appendix R fire safe shutdown analyses were reviewed thoroughly for the MUR power uprate and the conclusions in Section VII.1 apply: 1) existing operator actions are not affected; 2) no reduction in operator action time was identified; 3) no new operator actions were identified; and 4) no existing manual actions were automated.

   (2) The effect on the plant environment from the increased core power was evaluated, and it was concluded that the minor changes in temperature and pressure conditions will not prevent required post-fire operator manual actions that are identified in the Surry fire protection program from being performed within designated times.

   (3) Procedures and resources necessary for systems required to achieve and maintain safe-shutdown were reviewed and concluded to remain adequate for the MUR power uprate.

2. *The NRC staff notes that Attachment 5 to the LAR, Section VII.6.A.i, "Fire Protection Systems," states that "...The fire protection subsystems remain unchanged as a result of the MUR power uprate..." However, this section does not discuss the*

changes in the physical plant configuration related to the fire protection program or changes to the combustible loading at MUR power uprate conditions. Clarify whether this request involves any changes in plant configurations related to the fire protection program or changes to the combustible loading. If any, the staff requests the licensee to identify proposed changes and discuss the impact of these changes on the plant's compliance with the fire protection program licensing basis, 10 CFR50.48, or applicable portions of 10 CFR 50, Appendix R.

Dominion Response

No changes to the installed fire protection systems are required as a result of the MUR mechanical and electrical installation design modifications.

The electrical portion of the installation will affect the combustible loading in critical fire areas identified in Surry Technical Report EP-0012, "Combustible Loading Analysis," such as the Main Control Room and computer room where cables are installed in trays. This design change will add approximately 22.0 lbs of combustible material as a result of the addition of new cables to trays A30 and C30 in the Main Control Room and computer room area.

Cables are also being installed in the Cable Spreading Rooms, Mechanical Equipment Rooms and Turbine Building; however, they have no Appendix R impact since they are either not in a Fire Area that is quantitatively tracked or are in an area that has a qualitative analysis. The cables in these areas are being installed in steel conduit.

These modifications open and re-seal several existing penetrations and create new penetrations in several Appendix R fire barriers. New penetrations are being added for the cables going to the UFM panel from the spool pieces. Several floor penetrations from the Main Control Room are also being opened and re-sealed. Steel conduit is also being used.

The mechanical portion of the installations do not affect combustible loading. The physical requirements and response of the station fire protection systems remain unchanged.

The modifications implemented by these DCPs will not adversely impact the Station's design basis for compliance with Appendix R to 10 CFR 50. The mechanical installation will have a slight effect on one scenario assumed for the Surry B.5.b response, but the overall scenario and its goals are unchanged by these modifications.

Electrical Branch

1. *How does this increased loading affect the voltage drop through the service transformers and reserve station service transformers? Does it impact the Degraded Voltage Relay setting? How does this affect safety related loads when they start on the safety busses during an accident? How does this impact the load management discussed in the updated final safety analysis report (UFSAR) 8.4.1?*

   Dominion Response

   Station Service bus voltages are maintained via the main generator voltage regulator during unit operation in accordance with station operating procedures. The ability to maintain the specified bus voltages is not impacted. In addition, the Reserve Station Service transformers automatic load tap changers will continue to maintain their specified voltage at downstream buses.

   The increased loading of Station Service buses has minimal impact on the emergency buses. The increase in loading is due primarily to the Reactor Coolant Pump induction motors. Changes to other large induction motors are small. The voltage profile calculation simulates two types of transients; unit trip or unit accident. For these events, the transfer of Station Service buses to the Reserve Station Service transformers is delayed approximately 30 seconds after a turbine trip. For an accident scenario, the initial large motor starting load block is assumed to occur prior to the load transfer and is therefore unaffected. A calculation was performed to determine the impact on the voltage profile. As expected, the emergency bus voltages are decreased after the delayed transfer due to the higher loading. The resulting voltages are fully acceptable and do not approach the Degraded Voltage relay settings. The Reserve Station Service transformer automatic load tap changers will continue to correct voltage after the load transfer. No large challenging motor starting load blocks occur after the load transfer. The Degraded Voltage relay settings are unaffected.

   The voltage profile calculation performed to evaluate the impact of the Surry MUR also determined that the Reserve Station Service transformer loading remains well below the transformer ratings. The existing load shedding schemes continue to limit loading adequately. The transformers are rated for 30 MVA and the maximum calculated loading is 25.8 MVA. (The previous maximum was 24.3 MVA). The Station Service bus loading is also within the Station Service transformer and bus ratings.

2. *In Section V.1.D.i of Attachment 5 of the LAR, the licensee states that transmission system assessment included load flow studies of import/export system conditions and single-contingency, both normal and stressed, system conditions. Was this grid analysis performed for a dual unit trip after the increased loading of the power uprate? Furthermore, under this uprated conditions, can a fault in a Reserve Service transformer affect (trip) both units?*

Dominion Response

The Pennsylvania, New Jersey, Maryland Interconnection (PJM) grid analyses were performed per North American Electric Reliability Corporation (NERC) standards. N-1-1 assessments were performed as part of PJM's baseline assessments; however, re-dispatch is performed between each unit trip (e.g., in the model, trip Unit 1, re-dispatch the system, trip Unit 2 and vice versa). A simultaneous trip of both units was not performed, since it is not required per NERC standards. Likewise, the Surry UFSAR does not require evaluation of a dual unit trip.

The uprate does not impact the consequences of a fault in a Reserve Station Service transformer (RSST). During normal operation the Station Service buses are supplied from the Station Service transformers. The normal loading of the Reserve Station Service transformers and emergency buses are not impacted by the uprate. The Reserve Station Service transformers are protected by high speed differential protective relays that will limit the duration of the fault. RSST-A supplies emergency bus 1J and RSST-B supplies emergency bus 2H and RSST-C supplies emergency buses 1H and 2J. Accordingly, RSST-C has a higher probability of impacting both units since it supplies an emergency bus from both. The affected emergency buses are designed to separate from offsite power and align to the emergency diesel generators without tripping the unit(s).

3. *In Section V.1.F.i of Attachment 5 of the LAR, the licensee states that at uprate conditions the main generator for Surry Unit 1 and 2 will be capable of exporting 500 Mega Volt Ampere Reactive (MVAR) and importing approximately 430 MVAR. Also in Section V.1.D.i, the licensee states that Surry's generator output is limited to 400 MVARs out or 200 MVARs in, due to the 4 kV station service buses. If grid conditions are stressed, such that the 4 kV bus voltage is not the limiting factor, will the plant provide more reactive power to the grid (in excess of 400 MVARs)? If yes, was this factored in the stability analyses?*

Dominion Response

No. Due to procedural limitations, Surry Unit 1 and 2 will not provide reactive power in excess of 400 MVARs to the grid. Even though 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), Surry Unit 1 and 2 are only required by PJM, in our Interconnection Service Agreement, to operate at a power factor 0.95 leading to 0.90 lagging. The existing station operating procedures limit MVAR to 400 lagging and 200 leading, and main generator hydrogen pressure is currently operated at a maximum of 60 psig. The higher MVAR capability described would only be applicable after a turbine upgrade with implementation of changes necessary to operate the main generator hydrogen pressure at 75 psig. Operating procedure changes have not been processed for these future modifications. The current procedures properly reflect existing main generator limits and not the practical limits due to voltage constraints. The procedures permit operations staff to use the equipment for the full range of acceptable limits as long as other limits such as voltage are not exceeded. Future procedure changes will reflect the equipment limits again permitting operations staff maximum operational flexibility.

Stability analyses are developed based on the equipment ratings, maximum operating power (MW) limits, and applicable procedures associated with review and approval of the proposed generation changes. The stability analyses performed by the transmission provider are based on their internal guidance and NERC requirements.

4. *In section V.1.D.i of Attachment 5 of the LAR, the licensee states that the 941.7 Mega Volt Ampere (MVA) main generators have been replaced with 1055 MVA generators and associated exciters and voltage regulators. Additionally, the licensee states that the transmission system assessment did not require short circuit duty screening due to no changes in existing equipment. What affect does the main generators replacement have on the calculations performed for the grid analyses (short circuit duty screening)?*

<u>Dominion Response</u>

The 1055 MVA main generators have been in service for several years and are part of the model for the transmission system used for the analyses including short circuit duty. The available short circuit current from the generators is based on the generator's impedances and not on the power output level. The impact of the generator upgrade was evaluated at the time the generators were replaced. The MUR change has no impact on short circuit current.

5. *In section IV.1.A.vii of Attachment 5 of the LAR, the licensee states that the new worst-case reactor coolant pump (RCP) motor loads are larger than the RCP motor nameplate ratings. Furthermore, the licensee states that evaluations were conducted on the RCP motors to determine acceptability. Provide detailed discussion about the RCP motors worst-case loadings and the evaluation(s) performed to determine their acceptability. What is the worst-case voltage drop on the safety busses when the last RCP is started or operating at maximum load with grid at lowest allowable value? Discuss the affects of these conditions on the load management system described in UFSAR 8.4.1.*

<u>Dominion Response</u>

As described in a report provided by the Reactor Coolant Pump motor vendor, the temperature rises determined for the revised conditions for hot loop loading, cold loop loading, and for starting conditions still comply with the equipment specification requirements for the motors. Further, the changes to the thrust bearing loading for the MUR uprate conditions were shown to be insignificant. Therefore, the RCP motors are acceptable for operation at MUR power uprate conditions.

During unit startup, the Reactor Coolant Pump motors are supplied from the Reserve Station Service transformers. Each motor is supplied from a different transformer. As startup progresses and motors are added, each Reserve Station Service transformer will maintain voltage within the automatic load tap changer setpoint band regardless of loading on the other transformers. The starting current for the motors is based on the motor's impedance and not on the motor loading. Therefore, RCP motor starting voltages are unaffected by this change. Previous calculations have shown that emergency bus voltage will drop below the Degraded Voltage relay setting during Reactor Coolant Pump motor starting (but not below the Loss of Voltage relay setting). The Degraded Voltage relay timer setting is sufficiently long to permit motor starting and to permit the Reserve Station Service transformer automatic load tap changers to increase voltage after the motor start. The Degraded Voltage and Loss of Voltage relay settings permit RCP motor starting and are unaffected by the MUR change.

During unit startup, induction motor loading is gradually increased as loads are required. The Station Service buses are removed from the Reserve Station Service transformers at a low power level before bus loading is at the maximum level. Thus, Reserve Station Service loading during startup is not a concern.

The Station Service bus loading is within the Station Service transformer and bus ratings. The increased loading of Station Service buses has minimal impact on the emergency buses. The voltage profile calculation simulates two types of transients; unit trip or unit accident. For these events, the transfer of Station Service buses to the Reserve Station Service transformers is delayed approximately 30 seconds after a turbine trip. For an accident scenario, the initial large motor starting load block is assumed to occur prior to the load transfer and is therefore unaffected. A calculation was performed to determine the impact on the voltage profiles. As expected, the emergency bus voltages are decreased after the delayed transfer due to the higher loading. The resulting voltages are fully acceptable and do not approach the Degraded Voltage relay settings. The Reserve Station Service transformer automatic load tap changers will continue to correct voltage after the load transfer. No large, challenging, motor starting load blocks occur after the load transfer. The Degraded Voltage relay settings are unaffected. The voltage profile calculation performed to evaluate the impact of the Surry MUR also determined that the Reserve Station Service transformer loading remains well below the transformer ratings. The existing load shedding schemes continue to limit loading adequately. The transformers are rated for 30 MVA and the maximum calculated loading is 25.8 MVA. (The previous maximum calculated loading was 24.3 MVA.)

Attachment 2

REGULATORY COMMITMENT ASSOCIATED WITH THE RESPONSE TO THE NRC REQUEST FOR ADDITIONAL INFORMATION

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE LICENSE AMENDMENT REQUEST

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

REGULATORY COMMITMENT ASSOCIATED WITH THE RESPONSE TO THE NRC REQUEST FOR ADDITIONAL INFORMATION

The following list identifies only those actions committed to by SPS in this RAI response. Any other actions discussed in the submittal represent intended or planned actions described for information only and are not considered regulatory commitments.

COMMITMENT	SCHEDULED COMPLETION DATE (if required)
1. For the applicable UFSAR Chapter 14 events, Surry will re-analyze the transient consistent with Dominion's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).