VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 8, 2009

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

Serial No. 09-412
NLOS/ETS R1
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

In a letter dated March 26, 2009 (Serial No. 09-033), Dominion requested amendments to Operating Licenses NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. This measurement uncertainty recapture (MUR) power uprate License Amendment Request (LAR) would increase each unit's authorized core power level from 2893 megawatts thermal (MWt) to 2940 MWt, and make changes to Technical Specifications as necessary to support operation at the uprated power level. On June 2, 2009 the NRC sent a draft request for additional information (RAI) by e-mail. In a June 8, 2009 phone call, Dominion and the NRC staff discussed the draft RAI questions and Dominion agreed to provide a response by July 8, 2009. The formal RAI was received in a letter dated June 17, 2009. Attachment 1 provides the requested information.

During a review of the plant specific uncertainties calculations to address the RAI, Dominion identified errors in a steam moisture uncertainty calculation. The error, although small, affects the uncertainty values provided in Dominion's March 26, 2009 letters to support the license amendment request (Serial Nos. 09-033 and 09-033A). Therefore, revised pages for Attachments 1 and 5 of our submittal dated March 26, 2009 (Serial No. 09-033) and Attachments 2 and 3 of our submittal dated March 26, 2009 (Serial No. 09-033A) are included in Attachments 2 and 3, respectively. The revisions on each page are identified by lines in the right margin. Please replace the pages in these original documents with the revised pages to complete your review of the proposed license amendment.

The original Cameron Bounding Uncertainty Analysis Reports for Units 1 and 2, Attachments 2 and 3, of the March 26, 2009 letter (Serial No. 09-033A), are proprietary in nature and were withheld from public disclosure in accordance with 10 CFR 2.390. Consistent with the basis for withholding provided in the affidavit, Attachment 1 in Dominion's March 26, 2009 letter (Serial No. 09-033A), please withhold the revised pages included in Attachment 3 of this letter from public disclosure.

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The information provided in this letter does not affect the conclusion of the significant hazards consideration discussion provided in Dominion letter dated March 26, 2009 (Serial No. 09-033).

If you have any questions or require additional information, please contact Mr. Thomas Shaub at (804) 273-2763.

Sincerely,

J. Alan Price
Vice President - Nuclear Engineering

Attachments:

1. Response to RAI for License Amendment Request Measurement Uncertainty Recapture Power Uprate.
2. Revised Pages for License Amendment Request Measurement Uncertainty Recapture Power Uprate.
3. Revised Pages Cameron Bounding Uncertainty Analysis Reports for Units 1 and 2.

Commitments made in this letter:

1. Dominion will confirm that the variation in the flow normalization factors over a 48-hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the Ultrasonic Flow Meter flows during the 48-hour Completion Time prior to any such use above 2893 MWt.

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 8th day of July, 2009.

My Commission Expires: 4/30/13.

Notary Public

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My Commission Expires Apr 30, 2013

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ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

Background

By letter dated March 26, 2009 (Serial No. 09-033), (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090900055), Virginia Electric and Power Company (Dominion), submitted license amendment requests for North Anna Power Station, Units 1 and 2 (NAPS 1 and 2). The proposed amendment request would increase each unit's rated thermal power (RTP) level from 2893 megawatts thermal (MWt) to 2940 MWt, and make technical specification changes as necessary to support operation at the proposed uprated power level; an increase in RTP of approximately 1.6 percent. The Nuclear Regulatory Commission staff has reviewed the information the licensee submitted and determined that the following additional information is required to complete the evaluation.

To complete its review of the proposed modification and the Technical Specifications changes, the staff requests the licensee's response to the following:

NRC Questions

Containment and Ventilation

NRC Question 1

For the mass and energy release into containment resulting from the short-term loss-of-coolant-accident (LOCA), please explain why the lower power (and hence higher subcooling) of the MUR, relative to the power assumed in the current licensing basis analyses, does not result in more mass being discharged into the subcompartments and hence more conservative conditions than the existing analyses.

Dominion Response

Section II.2.31.2 in Attachment 5 of the license amendment request (LAR) stated: "The short-term LOCA mass and energy releases were generated at 102.2% of 2893 MWt." The LAR should have read as follows: "The short-term LOCA mass and energy releases were evaluated at 102.2% of 2893 MWt and determined to be conservative for the MUR uprate." This conclusion is consistent with Footnote 7 on Table II-2. A core power of 102.2% of 2893 MWt was reported in Section II.2.31.2 and Table II-2 based on the conclusion that the reactor coolant system (RCS) temperatures used in the licensing basis evaluations for short-term LOCA mass and energy releases remained bounding for RCS conditions at the current core power level of 2893 MWt and up to a maximum core power of 2956 MWt. Westinghouse used 2956 MWt as the bounding uprate power level (see Table 4.0-2 in Attachment 1 of the LAR). The basis for this conclusion is provided below in response to the NRC's question.

As core power increases with constant RCS flow, the cold leg temperature decreases and the hot leg temperature increases. Table 1 below compares the current full-power temperature conditions from the analysis basis with the MUR bounding conditions (Case 1 in Table 4.0-2 in Attachment 1 of the LAR) at the operating RCS average temperature of 580.8°F.

The current vessel outlet temperature of 615.6°F is less than the MUR uprate temperature of 616.3°F. Thus, the short-term LOCA mass and energy releases from a hot side break remains bounded by the existing analysis basis.

The current vessel inlet temperature of 546.0°F is greater than the MUR uprate temperature of 545.4°F. The effect of this temperature change on the short-term LOCA mass and energy releases was calculated to be insignificant. First, the increase in critical mass flux attributed to the temperature reduction was calculated to be less than 0.2% using the critical flow model from WCAP-8264-P-A (the North Anna licensing basis for short-term LOCA mass and energy releases). This change in itself is considered insignificant for subcompartment response. Second, the effect of the reduction in fluid enthalpy was calculated to determine the decrease in liquid flashing into a sealed subcompartment from a constant enthalpy process. Table 1 below summarizes the analysis input data and results. The change in enthalpy reduces the flashing mass such that the overall change in the short-term LOCA mass and energy releases would be a slight decrease (by ~0.04%).

In conclusion, reducing the cold leg temperature from 546.0°F to 545.4°F increases the break critical mass flux by less than 0.2%. The lower enthalpy of the colder water results in a smaller amount of liquid flashing. Overall, the effect on the flashing energy into a subcompartment is insignificant and the change in cold leg temperature would have no effect on the subcompartment pressure response. Thus, it is concluded that operation up to a maximum core/NSSS power of 2956/2968 MWt with the associated plant conditions in Table 4.0-2 of the LAR required no change to the short-term LOCA mass and energy releases. Operation at the MUR core power of 2940 MWt is bounded.

Table 1: Evaluation of Cold Leg Temperature Change on Short-term Mass and Energy Releases

	Reference Conditions	**MUR Case 1**	
NSSS Power, MWt	2910	2968	
Reactor Power, MWt	2898	2956	
RCS Tavg, °F	580.8	580.8	
Vessel Inlet, °F	546.0	545.4	
Vessel Outlet, °F	615.6	616.3	
Saturation Pressure at Vessel Inlet Temperature, psia (P_{sat})	1011.8	1006.8	ASME Steam Tables
Saturation Liquid Enthalpy at P_{sat}, Btu/lbm	544.4 (href)	543.6 (hmur)	ASME Steam Tables
Critical Mass Flux (G), lbm/sec-ft^2	27,763.3 (Gref)	27,809.7 (Gmur)	Note 1
Flow Ratio (FR)		1.00167	Gmur / Gref
Enthalpy Ratio (ER)		0.99790	Note 2
Total Effect = FR * ER		0.99957	

1) Modified Zaloudek critical mass flux is calculated using the NRC-approved methodology in WCAP-8264-P-A.
2) Enthalpy ratio = (h_{mur} – 180.2)/(h_{ref} – 180.2); where 180.2 Btu/lbm is the saturation liquid enthalpy at a containment pressure of 14.7 psia (upper bounding initial condition for North Anna's subatmospheric containment).

Reactor Systems

NRC Question 2

Describe and provide drawings of the location where the Ultrasonic Flow Meter (UFM) will be installed in the 3 main feedwater lines between the existing feedwater venturi flow meters and the main feedwater check valves.

Dominion Response

The basic configuration for the feedwater (FW) lines in both North Anna Units 1 and 2 are similar. The lines contain, in sequence, the FW flow venturis, a 90 degree pipe bend, the UFM spool piece metering sections and finally the main FW check valves. The location of the UFM spool piece metering sections can be specified relative to the centerline of the upstream 90 degree bend. The following table provides the actual distance downstream from the upstream bend.

Unit	FW Line	Distance downstream of the centerline of the feedwater line upstream of the pipe bend
NAPS U1	A	29 feet 3 inches
NAPS U1	B	14 feet 1 inches
NAPS U1	C	24 feet 0 inches
NAPS U2	A	18 feet 1 inches
NAPS U2	B	21 feet 7 inches
NAPS U2	C	28 feet 7 inches

Drawings showing the details of the UFM location for Unit 1 and 2 are enclosed.

NRC Question 3

In section I.1, (Application Attachment 5, Page 7), the description that spool pieces are installed well downstream of the existing feedwater flow venturis is unclear. Please quantify "well downstream" and justify that the spool pieces will have no impact on venturi performance.

Dominion Response

The FW piping is 16-inch, Schedule 80 pipe with a nominal inside diameter of 14.3 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 North Anna spool piece metering sections are installed a minimum of ten inside pipe diameters downstream of the FW flow venturis. The installation of the spool piece metering sections will create less than 0.015 psi of additional head loss in the feedwater system. Because the installed location of the spool piece metering sections is a minimum of 14 feet downstream of the centerline of the upstream FW pipe bends, the impact on the venturi is insignificant.

NRC Question 4

What are the instructions for transducer replacement?

Dominion Response

Station procedures 1-ICM-FW-UFM-001 and 2-ICM-FW-UFM-001 are under development with a current completion date of July 31, 2009. These procedures are based on Cameron Engineering Field Procedure 18 "Installation Procedure for In-Line Pushrod Transducer."

NRC Question 5

In Section I.1.G, (Attachment 5, Page 16), a completion time of 48 hours is proposed for operation in excess of 2893 MWth with the UFM not functional, provided that steady-state conditions persist throughout the 48-hour period. It is unclear how the "UFM not functional" is defined. Please describe the conditions that exist for a non-functional UFM.

Dominion Response

The North Anna 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 3.3.10 will provide the plant administrative controls for the Feedwater UFM Calorimetric and was included in Attachment 4 of Dominion letter Serial No. 09-033, dated March 26, 2009. Consistent with TRM 3.3.10, 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 from the Basis for TRM 3.3.10 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 TRM Basis is consistent with the plant design change that installed the feedwater UFM system and the control room annunciators. Section VII.2.B in Attachment 5 of Dominion letter Serial No. 09-033 dated March 26, 2009, describes the annunciators and that any UFM condition that increases feedwater flow uncertainty is considered a Feedwater Ultrasonic Flow Meter Failure alarm condition.

Steam Generator Tube Integrity and Chemical Engineering

NRC Question 6

Section II.3.2, "Auxiliary Equipment Design Transients" (Application Attachment 5, Page 47), indicates that the only auxiliary equipment design transients impacted by the power uprate are those associated with the reactor coolant system hot and cold leg temperatures. It is further stated that the existing auxiliary equipment design transients are conservative and bounding for the power uprate. Please discuss whether the analysis included changes in nitrogen-16 activity that would potentially effect letdown line decay time requirements.

Dominion Response

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.

Instrumentation and Controls

NRC Question 7

Section I.1.G "Completion Time and Technical Basis" (Application Attachment 5, Page 16) provides justification for the proposed 48 hours Allowed Outage Time (AOT) should the UFM be declared inoperable. The first bullet states "Alternate instrumentation accuracy due to nozzle fouling or transmitter drift will not significantly change over 48 hours." Was transmitter drift data used to support this conclusion? Please provide the calculated effect of the known transmitter drift on the power calorimetric calculation during the proposed AOT.

Dominion Response

The proposed Completion Time (CT) of 48 hours was based on engineering and industry experience with feedwater flow transmitters and consistent with the NRC approval of a 48-hour CT for Vogtle (referenced below). The original Foxboro feedwater flow transmitters for North Anna Units 1 and 2 were replaced with Rosemount transmitters in fall 2007 for Unit 1 and fall 2008 for Unit 2. Therefore, there is limited drift data available for these specific transmitters.

New calorimetric software to accommodate the leading edge flow meter (LEFM) is being installed in the Plant Computer System on both units. Once this software is in place, the feedwater venturi normalization factors (defined as feedwater mass flow from the LEFM divided by the calculated feedwater mass flow from the venturis and RTD's) can be calculated, tracked, and trended for all channels. Dominion will confirm that the variation in the flow normalization factors over a 48 hour period is negligible and that the normalized venturi flows are an acceptable surrogate for the LEFM flows during the 48-hour CT prior to any such use above 2893 MWt. Specifically, we will demonstrate that the effects of variability in the normalization factors between the UFM and venturi based flows, when statistically combined with other contributors to calorimetric uncertainty, does not produce an overall power uncertainty exceeding 0.37% (the margin between the analyzed limit of 102% of the current rating of 2893 MWt and the proposed MUR rating of 2940 MWt).

Dominion understands that, consistent with NRC practice to ensure completion of commitments prior to implementation of licensee amendments, this commitment and others may be included as license conditions with the MUR license amendment.

Reference:

Letter from Siva P. Lingham (NRC) to Mr. Tom E. Tynan (Vogtle Electric Generating Plant), *Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Recapture Power Uprate (TAC Nos. MD6625 and MD6626)*, February 27, 2008. (ML 080350347)

NRC Question 8

Section I.1.D.3.1 "Response to NRC Criterion 3" (Application Attachment 5, Page 12), Dominion Technical Report EE-0116 is referenced as the document that governs the combination of errors within instrument loops relative to setpoint determination. Please provide a copy of EE-0116. Has this report been previously reviewed by NRC? If so, please provide reference to that review.

Dominion Response

The Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) instrument setpoints and associated Allowable Values were derived using the setpoint methodology established in EE-0116, which is consistent with Methods 1 and 2 of the Instrumentation, Systems and Automation Society's (ISA) Standard ISA-R67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Revision 3 of EE-0116 determined the instrument setpoints using Methods 1 and 2 for both Surry and North Anna.

By letter dated September 19, 2007, (ADAMS Accession No. ML 072681096) Dominion provided Technical Report EE-0116, "Allowable Values for North Anna Improved Technical Specifications (ITS), Tables 3.3.1-1 and 3.3.2-1 and the Setting Limits for Surry Custom Technical Specifications (CTS) Sections 2.3 and 3.7," Rev. 3, dated September 2006, to support a license amendment to revise several setting limits for Surry Power Station. The methodology was reviewed and approved by the NRC staff and documented in NRC letter dated September 17, 2008, "Surry Power Station Units 1 and 2, Issuance of Amendments Regarding the Revision to Various Setting Limits and the Overtemperature ΔT/Overpower ΔT Time Constants (TAC Nos. MD6812 and 6813)," (ADAMS Accession No. ML 082250013).

Revision 4 of EE-0116, which was referenced in the March 26, 2009 submittal, incorporated changes to setpoints from plant changes and did not affect the setpoint methodology.

NRC Question 9

The steam enthalpy values presented in Item 7 of Table I-1 (Application, Attachment 5, Page 14), ER-646/Rev. 2 and ER-637/Rev. 3 for the NAPS 1 and 2 total thermal power uncertainty determination appear to be consistent with zero-moisture steam per ER-157P/Rev. 5. Please verify that the zero moisture steam condition is appropriate for NAPS 1 and 2.

Dominion Response

Table 1 in ER-157P, Revision 5, identifies two values for the uncertainty of steam enthalpy with the LEFM CheckPlus system. The bounding uncertainty of ± 0.22% power assumes an uncertainty of ± 0.25% in steam moisture about an assumed moisture of 0.25% (i.e., 100% uncertainty). Footnote 7 on Table 1 of ER-157P, Revision 5, supports a reduced uncertainty of 0.07% "if moisture carryover is known more accurately or if the steam is conservatively assumed to carry zero moisture in the heat balance calculation." Thus, there are two possibilities that can support a reduced uncertainty for steam enthalpy.

North Anna Units 1 and 2 include a steam moisture carryover (MCO) uncertainty of 0.08% moisture in the power calorimetric uncertainty calculation. Using this MCO uncertainty, a steam moisture contribution for power uncertainty of 0.013% rated thermal power (RTP) was developed. This value was combined with a steam pressure contribution to power uncertainty of 0.051% rated thermal power (RTP) to develop a plant-specific uncertainty for steam enthalpy of 0.053% RTP (presented as 0.05% in Table I-1 of the LAR). The steam moisture and pressure uncertainties are Items 20 and 21, respectively, in Table B-1 of ER-646, Revision 2 (Unit 1) and ER-637, Revision 3 (Unit 2) of Attachments 2 and 3, respectively, of our submittal dated March 26, 2009 (Serial No. 09-033A).

During the review of the plant-specific uncertainties calculation for this question, Dominion identified that the steam moisture uncertainty of 0.013% RTP that was used in the total thermal power uncertainty calculation was incorrect. The corrected steam moisture contribution to power uncertainty is 0.040% RTP. In response to this error, the plant-specific uncertainties calculation was revalidated completely and two small errors were identified that increase the Gains/Losses uncertainty from 0.087% RTP to 0.092% RTP [(Item 22 in Table B-1 of ER-646, Revision 2 (Unit 1) and ER-637, Revision 3 (Unit 2)].

The Cameron uncertainty calculations (ER-646 for Unit 1 and ER-637 for Unit 2) were revised with a steam moisture contribution to power uncertainty of 0.040% and a Gains/Losses uncertainty of 0.092%. The total thermal power uncertainty increased from 0.351% to 0.354% for Units 1 and 2. Both values remain less than the accident analysis allowance of 0.37%.

The impact on the contents of the LAR that was submitted by Dominion letter (Serial No. 09-033) dated March 26, 2009, is detailed below.

- The overall power level measurement uncertainty was reported as 0.35% at RTP on page 3 of Attachment 1 and on pages 8, 13, 14 and 21 of Attachment 5. This was a rounded value based on the original total power uncertainty of 0.351% RTP. The revised calculations produce a total power uncertainty of 0.354% RTP. Conservatively, this value would be rounded up to 0.36% RTP. The UFM-based calorimetric uncertainty remains less than the 0.37% RTP allowance in the deterministic accident analysis with a licensed power level of 2940 MWt. Therefore, the conclusion on page 21 of Attachment 5 that the deterministic accident analyses remain bounding is unaffected by the revised uncertainty calculations.
- The steam enthalpy uncertainty presented as Item 7 in Table I-1 on page 14 of Attachment 5 increases from 0.05% to 0.07% with the steam moisture contribution to power uncertainty of 0.040% RTP. The Gains/Losses uncertainty presented as Item 8 in Table I-1of Attachment 5 is unchanged from 0.09%.

The Cameron uncertainty calculations (ER-646, Revision 2 for Unit 1 and ER-637, Revision 3 for Unit 2) were provided as Attachments 2 and 3 of Dominion letter (Serial No. 09-033A) dated March 26, 2009. The thermal power uncertainty calculations have been revised to correct the non-conservative plant-specific uncertainties described above. Dominion is providing, in Attachment 3, the revised pages from ER-646, Revision 3 (Unit 1), and ER-637, Revision 4 (Unit 2).

<u>Electrical Engineering</u>

NRC Question 10

In Section III.2.A "Normal Operation" (Application Attachment 5, Page 57), the licensee states that the normal operation radiation dose levels increase as a result of the MUR power uprate for the reactor vessel excore neutron detectors and the qualified life of the excore detectors may be decreased. Furthermore, the licensee concludes that the preliminary results indicate no impact on radiation dose margin or qualified life of the excore detectors. Confirm that these calculations have been completed and that there is no impact on the radiation dose margin or qualified life of the excore detectors.

Dominion Response

As noted in our March 26, 2009 submittal (Serial No. 09-033), in Regulatory Commitment 10, Dominion committed to determine the EQ service life of the excore detectors prior to operating above 2893 MWt. The current schedule for completion of this commitment is September 30, 2009.

Dominion understands that, consistent with NRC practice to ensure completion of commitments prior to implementation of licensee amendments, that this commitment and others may be included as license conditions with the MUR license amendment.

NRC Question 11

In Section V.1.D.ii "Proposed New Generation Impact Analysis" (Application, Attachment 5, Page 101), the licensee states that the local generation study assessed station operation at maximum capability, and that the study identified no transmission deficiencies. Furthermore, the study indicated no decrement to system First Contingency Incremental Transfer Capability. The licensee states that in the summary section of the PJM impact studies, the maximum facility output is 945 MWe for Unit 1 and 938 MWe for Unit 2. Explain the results of this study and how the results are valid for the power uprate since Section V.1.F.i "Main Generator" (ADAMS Accession No. ML090900055, Attachment 5, Page 102), states that the output of the generators will be 980.5 MWe for Unit 1 and 972.9 MWe for Unit 2.

Dominion Response

Section V.1.F.i has been revised as follows to provide explanation of the PJM impact studies. The revisions are highlighted in bold type. Please use the following revised LAR Section V.1.F.i Main Generator to complete your review:

V.1.F.i Main Generator

Unit 1

The nameplate rating is 1105 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 475 MVARs out or 390 MVARs in, and maintain generator load and hydrogen pressure within the limits of the Generator Calculated Capability Curve with a generator rating of 1088.6 MVA. The main generator output at the current NSSS power level of 2905 MWt is 965 MWe. The anticipated main generator output is 980.5 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 980.5 MWe, the generator is capable of exporting 472.9 MVAR (lagging power factor of 0.900) and importing approximately 390 MVAR (leading power factor of 0.929). **However, the 980.5 MWe for Unit 1 is a gross MWe value and does not take into account the approximate 48 MWe of internal electrical loads the plant represents to the generator output for each unit. Subtracting the 48 MWe of internal electrical loads from Unit 1 heat balance value of 980.5 MWe yields a net maximum facility output of 932.5 MWe, which is below the 945 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 factor. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability **and the maximum facility output for Unit 1 is still bounded by the PJM studies.**

Unit 2

The Unit 2 main generator was replaced during the September 2008 outage. The exciter and voltage regulator were not replaced. The new nameplate rating is 1200 MVA (based on 75 psig hydrogen pressure), 0.900 power factor, and 22 kV. The generator is operated with restrictions not to exceed 444 MVARs out or 210 MVARs in and to maintain generator load and hydrogen pressure within the limits of Generator Calculated Capability Curve with a generator rating of 1088.6 MVA. The main generator output at the current NSSS power level of 2905 MWt is 965 MWe. The anticipated main generator output is 972.9 MWe based on the heat balance at MUR uprate conditions. The generator capability curve indicates that at 972.9 MWe, the generator is capable of exporting approximately 444 MVAR (lagging power factor of 0.910) and importing approximately 210 MVAR (leading power factor of 0.977). **However, the 972.9 MWe for Unit 2 is gross MWe value and does not take into account the approximate 48 MWe of internal electrical loads the plant represents**

to the generator output for each unit. Subtracting the 48 MWe of internal electrical loads from Unit 2 heat balance value of 972.9 MWe yields a net maximum facility output of 924.9 MWe, which is below the 938 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 factor. Therefore, the increase from the MUR power uprate remains below the main generator maximum capability **and the maximum facility output for Unit 2 is still bounded by the PJM studies.**

NRC Question 12

Provide the uprated loadings of the reserve station service transformers. Also, provide the ratings and the uprated loadings of the plant main transformers, and unit station service transformers.

Dominion Response

Reserve Station Service Transformers

The uprated loadings of the reserve station service transformers follow:

RSST A: 19.0 MVA
RSST B: 29.7 MVA
RSST C: 30.5 MVA

Main Transformers

The ratings of the main transformers are 1200 MVA. The uprated loadings of the main transformers are 1088.6 MVA minus the station service transformer loadings.

Station Service Transformers

The ratings of the station service transformers are 22.4 MVA. The uprated loadings of the station service transformers follow:

Unit 1	Unit 2
SST 1A: 19.3 MVA	SST 2A: 19.6 MVA
SST 1B: 19.2 MVA	SST 2B: 18.6 MVA
SST 1C: 18.8 MVA	SST 2C: 18.8 MVA

The transformer loadings provided in this response are based on the MUR uprated power conditions. Dominion is planning turbine replacement projects for North Anna Units 1 and 2, which will also affect the transformer loading values and the PJM Grid Stability Study. Similar engineering analysis will be performed to support the turbine replacements and the associated increased electrical generation. The turbine replacement project is independent of the MUR uprate. The supporting analysis for the turbine replacements is ongoing and scheduled to be completed in approximately six months for Unit 2 and two years for Unit 1. Based on the fabrication schedule for the turbines, the replacements are currently scheduled to be completed during refueling outages between 2010 and 2012.

Mechanical and Civil Engineering

NRC Question 13

In Section IV.1.A.v, "Balance-of-Plant Piping" (Application Attachment 5, Page 74), the licensee did not specify which Balance-of-Plant Piping systems were reviewed in support of the proposed power uprate. Please provide a list of these systems.

Dominion Response

The following North Anna Nuclear Power Station Units 1 and 2 balance of plant (BOP) and nuclear steam supply system (NSSS) interface piping systems were evaluated for MUR uprate conditions:

BOP Piping Systems

- Main Steam and Steam Dump System
- Moisture Separator & Reheater Drain System
- Feedwater System
- Condensate System
- Extraction Steam System
- High Pressure Feedwater Heater Drains System
- Low Pressure Feedwater Heater Drains System
- Gland Steam and Leak-off System
- Auxiliary Feedwater System
- Fuel Pool Cooling and Purification System
- Containment Spray (QSS and RSS) System
- Circulating Water System
- Auxiliary Steam System
- Chilled Water System
- Gaseous Waste System
- Liquid Waste System
- Service and Instrument Air System

NSSS Interface Piping Systems

- Chemical and Volume Control System
- Residual Heat Removal System
- Safety Injection System
- Pressurizer Spray System
- Pressurizer Safety Relief Valve and Power Operated Relief Valve Systems

Fire protection

NRC Question 14

In Section II.2.36, "Safe Shutdown Fire Analysis (Appendix R Report) – UFSAR 9.5.1," (Application, Attachment 5, Page 45), states that "...Operator actions in response to an Appendix R fire are not adversely impacted..." The staff requests the licensee to verify that (1) the MUR power uprate will not require any new operator actions, and (2) any effects from additional heat in the plant environment from the increased power will not interfere with existing operator manual actions being performed at their designated time and place.

Dominion Response

Section VII.1 of Attachment 5 of Dominion letter (Serial No. 09-033), dated March 26, 2009, 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 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.

The temperature and pressure parameters outside of containment will not be impacted by implementation of the MUR. Therefore, Appendix R manual actions outside containment are not affected by additional heat in the plant environment from the increased power.

A review of Fire Contingency Action (FCA) procedures 1/2-FCA-5 shows that there are no manual actions required inside containment to prevent or maintain safe shutdown. There is one manual action listed as a contingency in the event that less than one residual heat removal (RHR) pump is available: repair of RHR cable. This manual action would be required for cold shutdown approximately 55 hours after the fire initiating event. The fire extinguishment and hot shutdown time is approximately one hour. Therefore, this action inside containment is not affected by additional heat in the plant environment from the increased power.

Any effects from additional heat in the plant environment from the increased power will not interfere with existing operator manual actions being performed at their designated time and place.

NRC Question 15

Some plants credit aspects of their fire protection system for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If the NAPS 1 and 2 credits its fire protection system for other than fire protection activities, please identify the specific situations and discuss to what extent, if any, the MUR power uprate affects these "non-fire-protection" aspects of the plant fire protection system. If the NAPS 1 and 2 do not take such credit, please verify this as well.

Dominion Response

The North Anna Fire Protection System is not credited or required to mitigate the consequences of Design Basis Accidents. However as noted in UFSAR Chapter 9.5, 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 and for the Unit 1 and Unit 2 auxiliary feedwater systems. 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. The fire protection system's capacity remains adequate to provide secondary functions (i.e., backup water to the auxiliary feedwater pumps or makeup water to the spent fuel pool) at the uprated power. In addition, the Fire Protection System is used in B.5.b Mitigating strategies as a method to supply makeup water to various systems. The MUR has no impact on the ability of the Fire Protection System to meet the makeup requirements of the mitigating strategies.

<u>Accident Dose</u>

NRC Question 16

The discussion of the Waste Gas Decay Tank Rupture UFSAR 15.3.5 in Section II.2.19, (Application, Attachment 5, Page 36), refers to a calculation that indicates 1.6 rem at the exclusion area boundary. Although this value meets the Part 100 limit of 25 rem whole body which was originally used for tank rupture accidents, more recent guidance limits the dose to 500 mrem whole body or 100 mrem TEDE. The NRC staff notes that TS Section 5.5.11, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," item b, states that, "A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents ..." Provide additional information describing whether the Explosive Gas and Storage Tank Radioactivity Monitoring Program applies to the Waste Gas Decay Tank Rupture and if so why this program is not cited as the bases for making the determination that the MUR will not change the accident evaluation.

Dominion Response

Dominion agrees that the information in LAR section II.2.19 and TS 5.5.11 could be considered an inconsistency in the discussion of dose consequences from the rupture of a Waste Gas Decay Tank. As described in Section II.2.19, the dose consequence analysis of the Waste Gas Decay Tank Rupture results in 1.6 rem at the exclusion area boundary (EAB). This conservative analysis is part of the original plant licensing basis which will be maintained in the UFSAR. The Explosive Gas and Storage Tank Radioactivity Monitoring Program defined in TS 5.5.11 describes a station surveillance program that ensures that the quantity of radioactivity contained in the gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents. Discussion of this surveillance program will be included in LAR section II.2.19 as follows to provide a strong, defendable position that the MUR will have no impact on this accident.

From the March 26, 2009 submittal:

"II.2.19 Waste Gas Decay Tank Rupture - UFSAR 15.3.5

The waste gas decay tank rupture analysis was part of the original plant licensing basis. The analysis resulted in an exclusion area boundary whole body dose of 1.6 rem, which is reported in UFSAR Section 15.3.5 and compared to the 10 CFR 100 acceptance criterion. The 10 CFR 100 acceptance criterion for waste gas decay tank rupture exclusion area boundary whole body dose was 25 rem. Conservatisms in the radiological atmospheric dispersion factor (X/Q), dose conversion factors, and gap activities that were used in the analysis are such that the MUR power uprate impact on the waste gas decay tank rupture accident consequences is bounded."

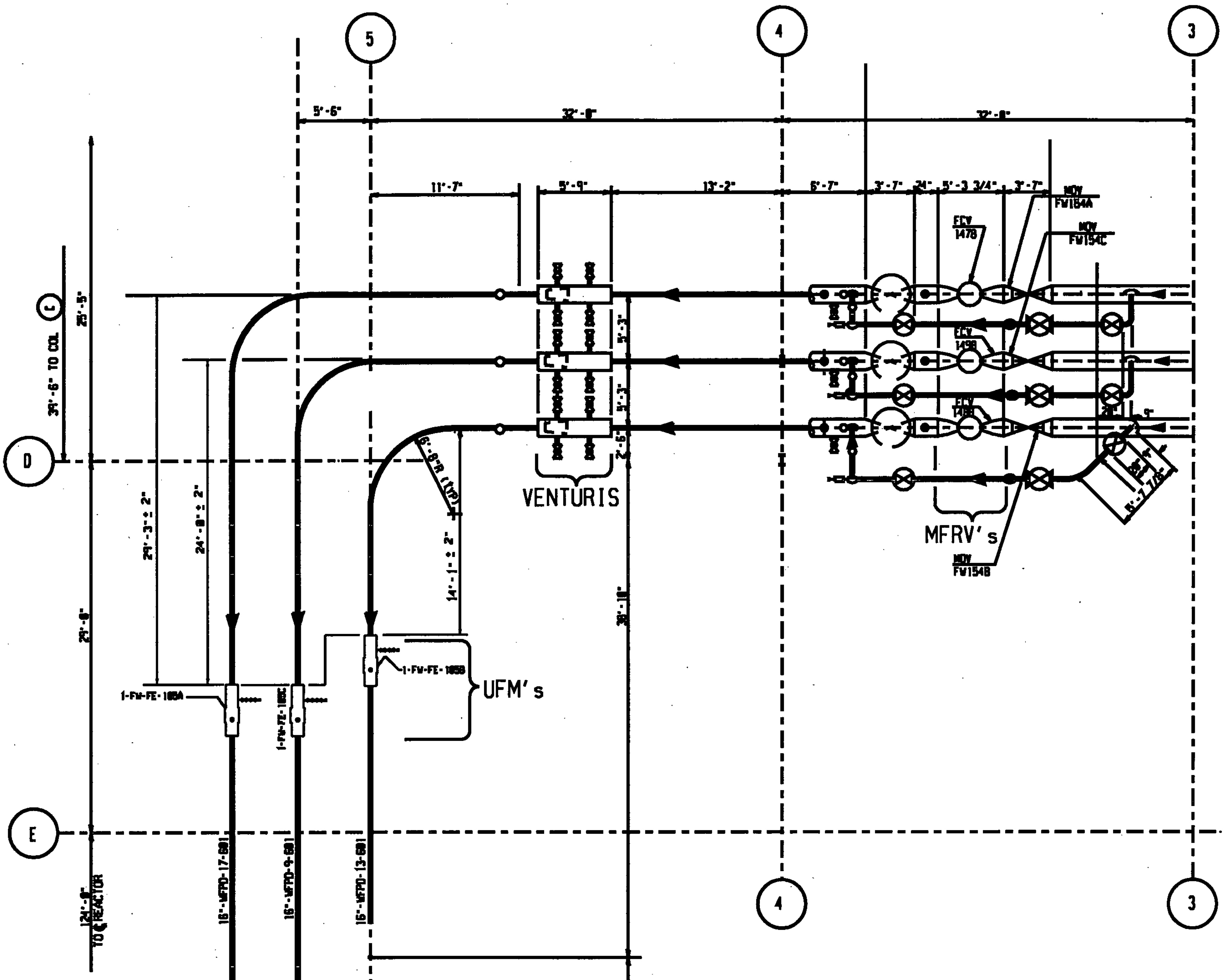
Please use the following revised LAR section II.2.19 to complete your review:

"II.2.19 Waste Gas Decay Tank Rupture - UFSAR 15.3.5

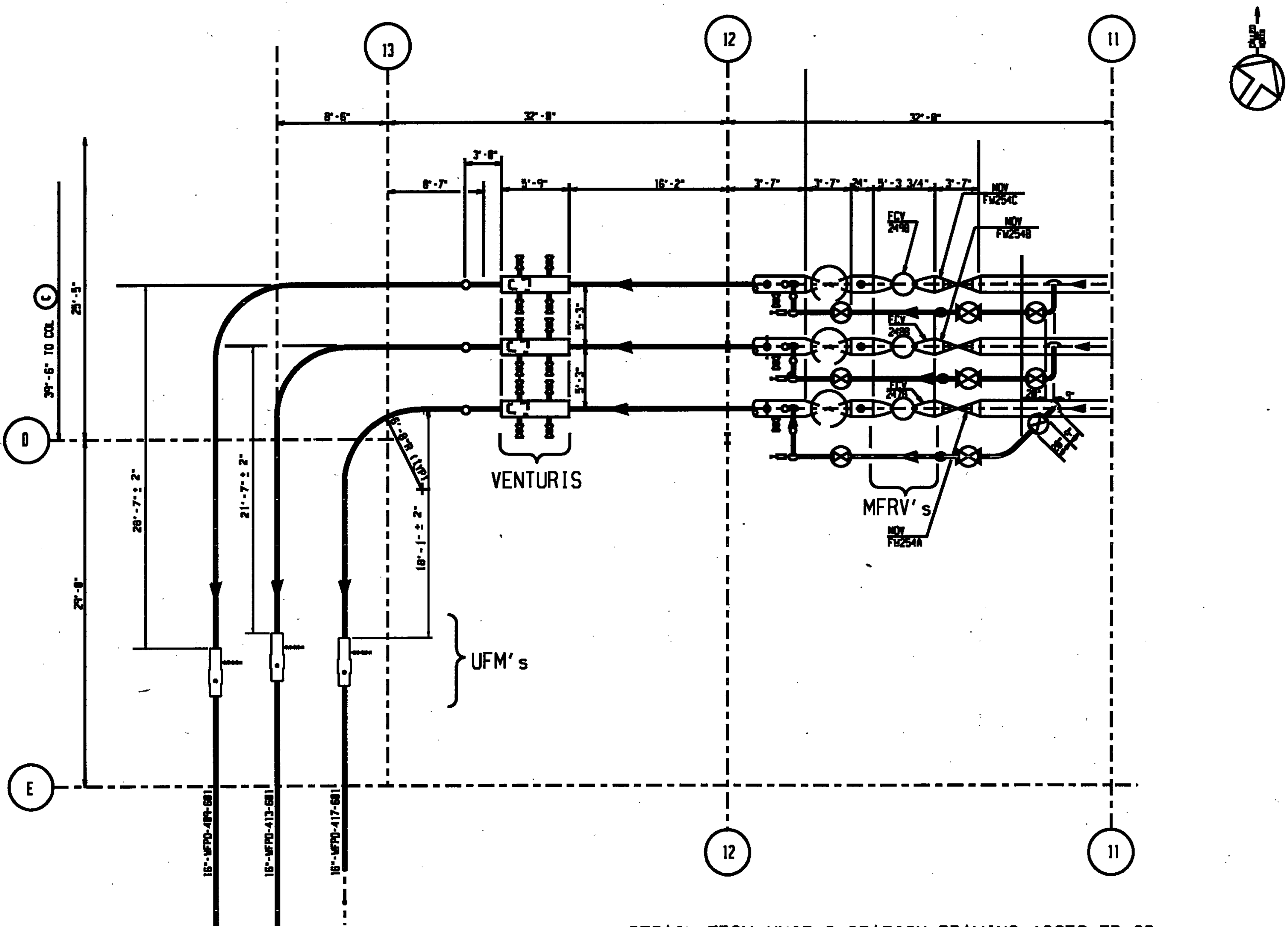
The Explosive Gas and Storage Tank Radioactivity Monitoring Program defined in TS 5.5.11 limits the quantity of radioactivity contained in a waste gas decay tank to less than an amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area in the event of an uncontrolled release of the tank's contents. The reported dose consequence of a waste gas decay tank rupture in UFSAR Section 15.3.5 is 1.6 rem which is part of the original plant licensing basis that complied with the 10 CFR 100 acceptance criterion of 25 rem to the exclusion area boundary. Due to the control of the tank's radioactive contents per the surveillance program defined in TS 5.5.11, the waste gas decay tank rupture accident is independent of power level. The surveillance program restricts the consequences of a tank rupture to < 0.5 rem, which is below the calculated value of 1.6 rem reported in UFSAR Section 15.3.5. Therefore, the MUR will have no impact on this accident."

Enclosure

Feedwater Line Drawings



DETAIL FROM UNIT 1 PROJECT DRAWING N-07012-1-1FP2C REV. 1



DETAIL FROM UNIT 2 STATION DRAWING 12050-FP-2B

ATTACHMENT 2

REVISED PAGES
LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

5 Pages

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2

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 changing current reactor core operations. 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 0.36% at RTP.

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 venturi-based feedwater flow indication and RTD temperature indication to perform the plant calorimetric measurement calculation. The currently installed venturi-based feedwater 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.

<u>Operating License - Maximum Power Level</u>

Paragraph 2.C(1), "Maximum Power Level," of the Unit 1 and Unit 2 Operating Licenses (NPF-4 and NPF-7 respectively) authorizes facility operation at a reactor core power level not in excess of 2893 megawatts

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 the 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 central processing unit (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 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 and a change in the way instrument uncertainty is combined for other parameters (e.g., steam temperature) results in a total uncertainty of 0.36% at RTP. This is more accurate than the nominal 2% RTP 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 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 NAPS 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 North Anna Units 1 and 2

The overall thermal power uncertainty using the UFM is 0.36% at RTP. The uncertainty calculations for North Anna 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 for determination of feedwater mass flow rate and enthalpy, the North Anna 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 North Anna uncertainties and those from ER-157P, Revision 5 are a result of plant-specific calculations and parameter uncertainties.

The uncertainty for transducer installation, as identified in Cameron Customer Information Bulletin CIB-125 (Reference I-10), has been included in the UFM uncertainty for North Anna Unit 1 (Reference I-6) and North Anna 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 North Anna 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	0.10%	0.15%	0.16%
3	Time Measurements Time of Flight Measurements Non-fluid delay	0.05%	0.15%	0.15%
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%	0.30%
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	0.07%	0.07%	0.07%
8	Gains/Losses	0.07%	0.09%	0.09%
9	Total Thermal Power Uncertainty	0.33%	0.36%	0.36%

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

flooding, station blackout, 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 NAPS MUR power uprate.

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

Analysis Scope	Core Power MWt	NSSS Power MWt(4)	Source
NSSS	2956(1)	2968	Design Parameters
Safety Analyses	2951(2)	2963	UFSAR Chapters 6 and 15
Statistical DNBR Events	2942.2(3)	2955	UFSAR Chapter 15

1. 102% of current analyzed core power of 2898 MWt
2. 102% of current RTP of 2893 MWt
3. 101.7% of current RTP of 2893 MWt
4. The analyses use 12 MWt for RCP net heat addition.

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 NAPS MUR power uprate, a core RTP of 2940 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 2951 MWt (102% of 2893 MWt) as the total core power, which leaves 11 MWt of margin to accommodate the power uncertainty. The 11 MWt is 0.37% of 2940 MWt. Since the power calorimetric uncertainty of 0.36% at RTP with the UFM is less than the accident analysis allowance of 0.37% with a 2940 MWt licensed power level, the deterministic accident analyses are bounding