UNITED STATES
NUCLEAR REGULATORY COMMISSION
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April 14, 2010

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

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2, REQUEST FOR ADDITIONAL INFORMATION: MEASUREMENT UNCERTAINTY RECAPTURE, (TAC NOS. ME3293 AND ME3294)

Dear Mr. Heacock:

By letter dated January 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100320264), U.S. Nuclear Regulatory Commission (NRC), the Virginia Electric and Power Company (the licensee), proposed to increase the authorized core power level of its Surry Power Station, Units 1 and 2, from 2546 megawatts thermal (MWth) to 2587 MWth. This increase in authorized power level (1.6-percent) is considered a measurement uncertainty recapture uprate.

The NRC staff has completed its preliminary review of this submittal and held a conference call on April 13, 2009. Based on the discussion, NRC needs additional information to complete the review. Our requests for additional information on the submittal are enclosed.

Please respond to this request for additional information no later than April 30, 2010.

Sincerely,

Jon Thompson for

Karen Cotton, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosure:
Request for Additional Information

Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

PROPOSED MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff has reviewed and evaluated the information provided by Virginia Electric and Power Company (licensee), in its letter dated January 27, 2010, which requested a license amendment to permit Surry Power Station, Units 1 and 2, to increase the maximum power level from 2546 MWt to 2587 MWt. The requested increase constitutes a Measurement Uncertainty Recapture (MUR) Power Uprate. The NRC staff requires additional information in order to complete its evaluation.

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." the "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 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.

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 (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?

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.

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?

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.

   (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.

2. Items Within the Reload Licensing Methodology Scope

   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.

3. Steam Line Break

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.

4. Control Rod Assembly Drop/Misalignment

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.

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.

6. Transducer Replacement

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.

7. Software

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

8. Self-Verification Feature

Explain the self-verification feature of the software.

9. Definition

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

10. Preventive Maintenance Program

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

11. Calibration and Maintenance

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

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.

13. Power Calorimetric

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

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 (χ/Q values) used in the steam generator tube rupture analysis. This section states that the control room (CR) and low population zone (LPZ) χ/Q values remain unchanged from the current licensing basis analysis 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 χ/Q values, which were also used for the MSLB accident dose assessment.

   NRC staff agrees that the LPZ χ/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 χ/Q values in the SE. Therefore, please provide a reference for and discussion of the CR χ/Q values, including confirmation that use of the CR χ/Q values for the MUR power uprate LAR remains unchanged from their use in the current licensing basis analysis.

Mechanical and Civil Engineering Branch

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) remains 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.

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.

   (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.

   (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.

   (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.

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.

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 NRC 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 CFR 50.48, or applicable portions of 10 CFR 50, Appendix R.

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 updated final safety analysis report (UFSAR) 8.4.1?

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 these uprated conditions, can a fault in a Reserve Service transformer affect (trip) both units?

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?

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)?

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.

Mr. David A. Heacock
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Glen Allen, VA 23060-6711

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Sincerely,

/RA by JThompson for/

Karen Cotton, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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