UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 18, 2009

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 REGARDING GENERIC LETTER 2004-02, "POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS" (TAC NOS. MC4722 AND MC4723)

Dear Mr. Heacock:

By letters dated February 29, 2008, and February 27, 2009, Virginia Electric and Power Company (the licensee) submitted supplemental responses to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," for Surry Power Station (Surry), Units 1 and 2 . The NRC staff has reviewed the licensee's submittals. The process involved detailed review by a team of 10 subject matter experts, with focus on the review areas described in the NRC's "Content Guide for Generic Letter 2004-02 Supplemental Responses" Agencywide Document Accession Management System (ADAMS) (Accession No. ML073110389). For its review, the NRC staff used the review guidance from several sources, including "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors dated March 28, 2008 (ADAMS Accession No. ML080230234), and the NRC's safety evaluation (SE) dated December 6, 2004 (ADAMS Accession No. ML043280641), from the Nuclear Energy Institute's May 28, 2004 document, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (ADAMS Accession No. ML041550661). The review process also included a separate review of the licensee's submittal informed by inputs from the subject matter experts that focused on whether the licensee has demonstrated overall that its corrective actions for GL 2004-02 are adequate.

Based on these reviews, the NRC staff has concluded that additional information is needed to conclude that GL 2004-02 has been satisfactorily addressed for Surry. The enclosed document forwards the request for additional information (RAI).

The NRC staff requests that the licensee respond to this RAI within 90 days. However, the NRC staff wishes to only receive one response letter, with the exception of RAI No. 11, as discussed below. If the licensee concludes that more than 90 days are needed to respond to the RAI, the licensee should request additional time, including a basis for why such time is needed.

As part of the written response to the additional RAI, the NRC staff requests that you include a safety case. This safety case should describe, in an overall or holistic manner, how the measures credited in the Surry licensing basis demonstrate compliance with the applicable NRC regulations as discussed in GL 2004-02. This safety case should inform your approach to responding to the RAI, as well as the NRC staff's review of the RAI responses. As appropriate, it may describe how you have reached compliance even in the presence of remaining uncertainties. The NRC staff sees the safety case as informing, not replacing, responses to the RAI.

Regarding RAI No.11, the NRC staff considers in-vessel downstream effects to not be fully addressed at Surry as well as at other pressurized-water reactors. The licensee's submittal refers to draft topical report WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final SE for WCAP-16793-NP. The licensee may demonstrate that in-vessel downstream effects issues are resolved for Surry by showing that the licensee's plant conditions are bounded by the final approved WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing the conditions and limitations in the final SE. The licensee may also resolve this item by demonstrating without reference to WCAP-16793-NP or the NRC staff SE that in-vessel downstream effects have been addressed at Surry. In any event, the licensee should report how it has addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793-NP.

In addition, given that the draft WCAP-16793 Revision 1 is now available to licensees and it is the best available industry information of which the NRC staff is aware regarding plant susceptibility to this issue, please review Surry 1 and 2 against the draft WCAP-16793 criteria if you have not already done so. Please inform the NRC staff of your planned actions if the results of the evaluation show that Surry 1 and 2 are not bounded by the draft WCAP-16793. White the NRC staff has not yes issued a final safety evaluation for WCAP-16793, it is prudent for licensees, who plan to rely on the topical report to evaluate their plant-specific conditions against the draft report.

Sincerely,

R Martin
For John Stang, Senior 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:
As stated

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION REGARDING

SUPPLEMENTAL RESPONSES TO GENERIC LETTER 2004-02

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1. Please describe how much aluminum surface area (from the reactor vessel insulation) would be exposed for a reactor coolant system (RCS) loop break at a reactor vessel nozzle. Please explain whether your chemical effects evaluation considered exposure of this material and whether a break at this location is potentially limiting with respect to potential for sump strainer clogging.

2. Please describe the construction details for the asbestos and asbestos/Cal-Sil insulation at Surry and provide results of evaluation of the similarity of these materials in the plant to the Cal-Sil material whose testing formed the basis for the zone of influence (ZOI) value of 5.45D that is referenced in Nuclear Energy Institute 04-07, and the corresponding NRC safety evaluation (SE), cited in the February 29, 2008, supplemental response as applicable to the Surry Power Station's asbestos and asbestos/Cal-Sil. Please also explain how the base material, jacketing, and banding for the material in the plant are similar to the properties of the tested material used to derive a 5.45 ZOI.

3. Please provide additional information that justifies the temperature/viscosity extrapolation of data from test temperatures to predicted loss-of-coolant accident (LOCA) temperatures. Based on recent review of Rig 33 head loss traces for North Anna during the chemical effects audit of that plant, the NRC staff believes that there may not have been "sudden" decreases in measured head loss, but there were anomalous observations of fairly large and relatively fast head loss decreases for qualification tests and other non-qualification tests for North Anna. Flow sweeps were done for some of the tests that seemed to indicate that boreholes did not have a significant influence on the temperature scaling. However, the NRC staff does not consider this information sufficient to conclude that there were no signs of potential bed degradation. Please provide results of evaluation of the cause of the decreases in head loss that occurred during testing.

4. Please provide an evaluation similar to that provided for North Anna to show that the results of both Rig 33 and Rig 89 tests, and the magnitude of plant-specific conservatisms for Surry, ensure that the strainers will function under design conditions.

5. The minimum strainer submergence was the same for both large-break and small-break loss-of-coolant LOCAs. It was not clear what sources were credited for the minimum level calculation. Please state whether the accumulators are credited for small break LOCA sump level calculations. If the accumulators are credited for small breaks, provide justification for

this assumption, or provide the minimum water level if no accumulator volume is credited. Please state whether any RCS volume is credited for the minimum water level calculation. If RCS volume is credited, please provide the volume credited and the assumptions and bases for the credited volume.

6. Please provide an evaluation of the head loss fluctuations that occurred during the low-head safety injection (LHSI) Rig 89 testing between the 7th and 10th aluminum additions. Also, please explain why these fluctuations do not invalidate any viscosity corrections imposed on the test data.

7. The licensee's February 29, 2008, supplemental response indicated that the methodology for the Surry net positive suction head (NPSH) calculation was similar to that reviewed for North Anna during the GSI-191 audit for that plant. However, plant-specific differences and results for Surry were not provided in the supplemental response as requested in the NRC staff's content guide. Please provide the following information requested in the content guide. The responses may be in terms of stating that the same approach was used as for North Anna, or of describing any differences from the North Anna approach, which the NRC staff has already reviewed.

    a. a description of the methodology for computing the maximum flows for the LHSI and RS pumps
    b. the basis for the required NPSH values, e.g., three percent head drop or other criterion.
    c. a description of how friction and other flow losses are accounted for.
    d. a description of the single failure assumptions relevant to pump operation and sump performance that were considered in the NPSH calculation
    e. assumptions that are included in the analysis to ensure a minimum (conservative) water level is used in determining NPSH margin
    f. a description of whether and how the following volumes have been accounted for in pool level calculations: empty spray pipe, water droplets, condensation and holdup on horizontal and vertical surfaces. If any are not accounted for, explain why.
    g. assumptions (and their bases) as to what equipment will displace water resulting in higher pool level

8. Please provide a description of how permanent plant changes inside containment are programmatically controlled so as to not change the analytical assumptions and numerical inputs of the licensee analyses supporting the conclusion that the reactor plant remains in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 and related regulatory requirements.

9. Please provide a description of how maintenance activities, including associated temporary changes, that could affect the licensee's analytical assumptions and numerical inputs of the licensee's analyses relating to its resolution of sump performance issues, are assessed and managed in accordance with the Maintenance Rule, 10 CFR 50.65.

10. Page 56 of 64 of the February 29, 2008, supplemental response indicates that the numerical data relating to the structural qualification of the replacement strainers is contained in two AECL Seismic Analysis Reports for Surry Power Station. In accordance

with the second bullet in Section 3.k of the Revised Content Guide for GL 2004-02 Supplemental Responses, please provide and summarize, in tabular form, the design margins for the strainer components analyzed for structural adequacy.

11. The NRC staff considers in-vessel downstream effects to not be fully addressed at Surry as well as at other PWRs. The licensee's submittal refers to draft WCAP-16793-NP-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final SE for WCAP-16793-NP. The licensee may demonstrate that in-vessel downstream effects issues are resolved for Surry by showing that the licensee's plant conditions are bounded by the final WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing the conditions and limitations in the final SE. The licensee may also resolve this item by demonstrating without reference to WCAP-16793-NP or the NRC staff SE that in-vessel downstream effects have been addressed at Surry. In any event, the licensee should report how it has addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793-NP.

12. The licensee's letter dated February 27, 2009, states (page 39 of 43) that "a review of ICET results indicated minimal transport of aluminum surfaces sprayed for four hours, therefore it can be concluded that the aluminum released to the sump in the short term originates solely from submerged aluminum." On that basis, the licensee concluded that potential chemical effects during the first four hours after a LOCA would be insignificant. The NRC staff agrees that some corrosion product was retained on the sprayed aluminum samples in the relevant ICET tests. The NRC staff, however, does not understand how this observation leads to the conclusion that the aluminum originated solely from the submerged aluminum coupons since the NRC staff is not aware of how the measured dissolved aluminum concentrations could be apportioned into contributions from submerged and sprayed samples. Please provide the basis for this conclusion or provide alternate reasons (e.g., aluminum is more soluble at the higher pool temperatures present in the short-term following a LOCA) why the potential chemical effects are initially expected to be insignificant.

13. Please describe how transported debris was assumed to be apportioned between the recirculation spray and LHSI strainers, and provide the basis for considering dual-train operation of the LHSI system to be bounded by single-train operation. With two LHSI pumps running, the total debris accumulating on the LHSI strainer would be greater, which in turn could result in an increased head loss that exceeds the conservatism associated with the NPSH evaluation for the single-train case under clean strainer conditions.

credited in the Surry licensing basis demonstrate compliance with the applicable NRC regulations as discussed in GL 2004-02. This safety case should inform your approach to responding to the RAI, as well as the NRC staff's review of the RAI responses. As appropriate, it may describe how you have reached compliance even in the presence of remaining uncertainties. The NRC staff sees the safety case as informing, not replacing, responses to the RAI.

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Sincerely,

/RA BMartin for/
John Stang, Senior 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:
As stated

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