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

April 1, 2020

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2; AND SURRY POWER STATION, UNIT NOS. 1 AND 2 - RE: REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGY (EPID L-2018-LLA-0195 AND L-2018-LLA-0215)

Dear Mr. Stoddard:

By letters dated July 12, 2018 and July 31, 2018 (Agencywide Documents Access and Management System Accession Nos. ML18198A118 and ML18218A170), Virginia Electric Power Company (the licensee) submitted license amendment requests (LARs) for the North Anna power Station (NAPS), Units 1 and 2, and for the Surry Power Station (SPS), Units 1 and 2, respectively, requesting approval to implement a fuel vendor-independent evaluation model for analyzing hypothetical small break loss-of-coolant accidents.

As part of its review of the LARs, staff from the U. S. Nuclear Regulatory Commission (NRC) conducted an audit at the licensee corporate offices in Glen Allen, Virginia, from January 22-24, 2020.

As a result of its review and the interactions at the audit, the NRC staff has determined that the responses to the questions contained in the attached request for additional information are needed to complete our evaluation. We request that you provide a written response to this request within 30 days.

~~Enclosure 1 to this letter contains Proprietary information. When separated from Enclosure 1, this document is DECONTROLLED~~

If you have any questions, please contact me at (301) 415-2481, or via e-mail at ed.miller@nrc.gov.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch 2-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338, 50-339, 50-280, and 50-281

Enclosures:
1) Proprietary RAIs
2) Redacted RAIs

cc: Listserv

REQUEST FOR ADDITIONAL INFORMATION RELATED TO

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS 1 AND 2,

NORTH ANNA POWER STATION UNITS 1 AND 2

REQUEST TO IMPLEMENT FUEL VENDOR INDEPENDENT EVALUATION MODELS

DOCKET NOS. 50-280, 50-281, 50-338, AND 50-339

Information Sensitivity:

~~**The following document contains proprietary information. The proprietary information is withheld from public disclosure pursuant to 10 CFR 2.390. Proprietary information within the document is marked between double square brackets.**~~

~~**[[**This sentence is an example of the proprietary designation.**]]**~~

RAI 2 S1:

The response to request for additional information (RAI) 2 refers to calculated results from the FVI-SBLOCA [fuel-vendor independent small-break loss-of-coolant accident] and ASTRUM [Automated Statistical Treatment of Uncertainty Method] large-break loss-of-coolant accident methods as demonstrating that breaks in a size range between 10% of the cold leg cross-sectional area and 1.0 ft^2 are adequately addressed. However, the response appears to be based on extrapolation of FVI-SBLOCA and ASTRUM results into a range of the break spectrum (i.e., from approximately 0.4125 – 1.0 ft^2) where no calculations for North Anna or Surry have been reported with either evaluation model.

Furthermore, the U.S. Nuclear Regulatory Commission (NRC) staff observed in Section 15.3.1.5.1 of the North Anna Updated Final Safety Analysis Report (UFSAR) that “The NOTRUMP computer code is used for loss-of-coolant accidents due to small breaks less than one square foot.” A similar description exists in Section 14.5.2.2 of the Surry UFSAR. The UFSAR descriptions reviewed by the staff do not appear to define any portion of the postulated LOCA break size range as inherently non-limiting.

Paragraph 10 CFR 50.46(a)(1)(i) requires that “ECCS [Emergency Core Cooling System] cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." As such, please address the following RAIs:

(a) Compare the proposed range of small breaks for the FVI-SBLOCA methodology to the analyzed range of breaks for the current SBLOCA evaluation model. Please provide justification if adoption of the FVI SBLOCA methodology would result in a reduction to the analyzed break spectrum as compared to the current evaluation model.

(b) Considering that the predicted limiting break size may in general be a function of, among other things, the evaluation model being used, please provide any evidence, such as calculated results using the FVI-SBLOCA and ASTRUM methods, demonstrating that these evaluation models will not predict limiting results for the LOCA event in the range of break sizes between 0.4125 – 1.0 ft^2.

RAI 6 S1:

The licensee's response to RAI 6 provides a qualitative, historical review of background material concerning the analysis of reactor coolant pump trip timing for the SBLOCA event. Much of the historical analyses and derivative insights discussed therein originated in response to the 1979 accident at Three Mile Island (TMI), Unit 2. As discussed during the regulatory audit in October 2018, the post-TMI analysis relied upon computer codes developed during the 1960s and 70s with significantly simpler modeling practices than modern codes (e.g., 10-20 fluid nodes, simplified field equations). The post-TMI analysis also focused upon smaller break sizes (e.g., 2-4 inches), as opposed to the larger range of small breaks discussed in RAI 6 (i.e., 5 inches and larger) that contemporary analyses show have the potential to be limiting for many Pressurized Water Reactors (PWRs) **[[**[redacted] The licensee's response to RAI 6 did not describe sensitivity calculations applicable to North Anna or Surry in the range of reduced reactor coolant pump trip delay times and break sizes of interest to RAI 6.

Based upon its review of the licensee's response to RAI 6, the NRC staff concluded that the concerns expressed in RAI 6 that trip times less than 5 minutes could be both (1) more limiting than the cases analyzed by the licensee for break sizes 5 inches and larger and (2) more likely than the cases analyzed by the licensee had not been adequately addressed. To probe the significance of the issue, the NRC staff performed preliminary sensitivity studies using the TRACE thermal-hydraulic code that considered **[[**[redacted]**]]**.

During a regulatory audit held on January 22-24, 2020, the NRC staff audited a calculation report containing sensitivity analyses for reduced reactor coolant pump trip times using the EMF-2328 methodology that appeared to show similar results to the staff's calculations using TRACE. The EMF-2328 sensitivity results, which assumed **[[**[redacted]

[redacted]]

Paragraph 10 CFR 50.46(a)(1)(i) requires that “ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.” As described above, the sensitivity studies illustrate that such assurance has not been provided, because the presently limiting break size indicates the potential to return more severe results when the reactor coolant pump (RCP) trip timing sensitivity is also considered. As such, please provide the following additional information:

Adequately address the potential for **[[** [redacted] **]]** for North Anna and Surry that has the potential both to produce more severe consequences and to be more likely than the cases analyzed by the licensee [redacted] **]**.

As applicable, **[[** [redacted] **]]**.

Identify the value(s) of Reactor Coolant System (RCS) subcooling margin at which **[[** [redacted] **]]** for North Anna and Surry.

<u>RAI 7 S1:</u>

The response to RAI 7 indicates the results of the refueling water storage tank (RWST) drain down sensitivity study support that the analyses presented in ANP-3467P and ANP-3676P remain bounding. The response to RAI 7 also indicates that the **[[** [redacted] **]]**, that this value was assessed as bounding based on **[[** [redacted] **]]**. The response did not provide sufficient detail to conclude that the supporting analyses establish **[[** [redacted]

Paragraph 10 CFR 50.46(a)(1)(i) requires that ECCS cooling performance must be “calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.” As such, please address the following RAIs for both North Anna and Surry:

(a) Identify the break sizes considered in the RWST drain down sensitivity analyses and provide justification for the adequacy of the assessed break sizes.

(b) **[[** [redacted] **]]**.

(c) Supplement Tables 4-2 of both ANP-3467P and ANP-3676P by indicating **[[**[redacted]**]]**.

(d) Provide adequate technical basis for the conclusion that a post-RWST drain down **[[**[redacted]**]]** is an appropriately conservative **[[**[redacted]**]]**.

RAI 8 S1:

The NRC staff requested in part b. that the licensee estimate the observed change in peak cladding temperature associated with an S-RELAP5 code modification autonomously implemented by Framatome following the NRC staff's review and approval of the SBLOCA evaluation model described in EMF-2328(P)(A).

The requirements in Appendix K to 10 CFR 50 reflect the importance of performing comparisons of evaluation model predictions against relevant test data. The assessment of the EMF-2328 evaluation model against test data, which constitutes part of the NRC staff's basis for finding the evaluation model acceptable, is specifically discussed in Section 4.5 of the NRC staff's safety evaluation of Revision 0 and Section 5.3 of the NRC staff's safety evaluation of Supplement 1.

Confirmation of the impact of the autonomously implemented code modification on the calculated peak cladding temperature and other relevant figure of merit specified in 10 CFR 50.46(b) is necessary to confirm whether (1) the existing evaluation model assessment remains valid or (2) a new assessment is necessary with the modified evaluation model the licensee proposes to apply to North Anna and Surry.

Therefore, please provide the following information:

(a) Please provide a valid estimate of the magnitude of the impact on peak cladding temperature and other relevant figures of merit specified in 10 CFR 50.46(b) that is associated with the S-RELAP5 modification Framatome autonomously implemented following the NRC staff's review and approval of the EMF-2328(P)(A) evaluation model.

(b) Please justify that the autonomously implemented code modification does not adversely affect previously reviewed assessments of the EMF-2328 evaluation model in both Revision 0 and Supplement 1. Please include available supporting evidence, such as calculated results from a vendor continuity of assessment evaluation, which demonstrates the impact of reanalyzing assessment cases with the modified code version.

(c) Please clarify **[**[redacted]**]]**

RAI 11 S1

The response to RAI 11 appropriately identified **[[** [redacted] Calculation of **[[** [redacted] **]]** is an essential element of adequately predicting the termination of the cladding heatup, and hence the peak cladding temperature and maximum local oxidation; therefore, **[[** [redacted] **]]** must be taken into account when demonstrating that the acceptance criteria contained in 10 CFR 50.46(b)(1) and (2) are satisfied. **[** [redacted] **]]**. Confirm that the response to RAI 11 insofar as it pertains to **[[** [redacted] If not, please provide a detailed description and justification for the **[[** [redacted] **]]** in accordance with Section II, "Required Documentation," of Appendix K to 10 CFR 50.

RAI 12.A S1

The response to RAI 12.a identified **[[** [redacted] **]]** parameters that would be considered in determining whether a given analysis performed with the FVI SBLOCA methodology remains valid for an upcoming fuel cycle, or whether a new analysis must be performed:

- **[[** [redacted] **]]**

According to the response to RAI 12.a, other fuel-related parameters associated with **[[** [redacted] **]]** would not be considered in the applicability determination.

Due to the non-linear impacts of fuel behavior on the LOCA event (e.g., with respect to cladding deformation and rupture), development of a set of acceptance criteria for continued analysis applicability that is both simplified and universal may be challenging. For instance, although the **[[** [redacted] **]]** presented in response to RAI 19 may apply to existing plant conditions, it does not necessarily apply to other conditions (e.g., more severe cladding temperature transients).

Moreover, the responses to RAI 12 and other questions (e.g., RAIs 20, 21, and 22) did not offer adequate validation that fuel parameters other than those considered by the licensee would not affect the continued applicability of an analysis performed using the FVI SBLOCA methodology to future fuel cycles.

Based upon the information in the RAI responses and previously docketed submittals, the NRC staff's review needs additional information to confirm that the **[[** [redacted] **]]** criteria proposed by the licensee would be sufficient, in general, to assure an acceptable determination of the peak cladding temperature and other figures of merit specified in 10 CFR 50.46(b) on a cycle-specific basis.

Therefore, to assure that the proposed FVI SBLOCA methodology is capable of acceptably calculating figures of merit for comparison with the acceptance criteria specified in 10 CFR 50.46(b), please (1) provide adequate evidence demonstrating that the proposed criteria for determining the continued applicability of an analysis on a cycle-specific basis are valid or (2) propose a modified set of criteria or methodology for making the determination that is supported by adequate justification.

RAI 12.B S1

The response to RAI 12.b appears to identify **[[**[redacted]**]]** criteria for determining whether the FVI SBLOCA methodology may be applied to a given fuel design:

- **[[**[redacted]

The response to RAI 17 further appears to add an additional criterion that **[[**[redacted]**]]**:

- Fuel has the same lattice structure as the current resident fuel

The NRC staff's review of these criteria identified the potential for applications of the FVI SBLOCA methodology to fuel designs with properties and features that extend beyond the set of characteristics explicitly considered in the NRC staff's review of the Framatome codes supporting the methodology. For instance, EMF-92-116(P)(A) indicates that RODEX2 may be used to model fuel for U.S. PWRs using uranium dioxide and urania-gadolinia pellets, and Zircaloy-4 or M5 cladding. The use of the RODEX2 code for fuel incorporating other types of cladding materials, burnable poisons, and other design features not currently used in Framatome fuels, but which may be incorporated into other vendors' fuel designs, has not been previously reviewed. Existing reviews of the RODEX2 code also did not previously review the impacts of many novel design features being considered for future fuel rods, such as advanced cladding alloys and coatings; different fuel materials, enrichments, geometries, dopants, and coatings; and advanced fabrication techniques. The NRC staff's safety evaluation on EMF-92-116(P)(A) further assessed the RODEX2 code only up to Framatome's currently licensed maximum burnup of 62 GWd/MTU. Although the criteria proposed by the licensee in response to RAI 12.b would apparently allow the previously reviewed application scope for the RODEX2 code to be exceeded, adequate justification for application of RODEX2 to such expanded applications has not been provided. Furthermore, note that this discussion applies not only to **[[**[redacted]**]]**.

Additionally, the licensee's response to RAI 13 did not provide assurance that a sufficiently robust administratively controlled process exists to incorporate potentially incomplete data for other fuel cladding types and other design features into analyses performed using the FVI SBLOCA methodology.

Finally, regarding application of the FVI SBLOCA methodology to all fuel designs clad with a zirconium-based alloy, the NRC staff noted that a comparison of the response to RAI 21 and results presented in ANP-3315P indicates [[redacted]]. Additional discussion of the potential for fuels clad with different zirconium-based alloys to behave differently during an SBLOCA event is presented below in RAI 18 S1.

To assure that the proposed FVI SBLOCA methodology is capable of acceptably calculating figures of merit for comparison with the acceptance criteria specified in 10 CFR 50.46(b), please list the criteria that would be used to determine applicability of the FVI SBLOCA methodology to a specific fuel design. Furthermore, please (1) provide adequate justification that the proposed criteria for determining the applicability of the FVI SBLOCA methodology to a specific fuel design are valid or (2) propose a modified set of criteria or methodology for making the determination that is supported by adequate justification.

RAI 13 S1

The response to RAI 13 discusses an "established process" for determining whether the characteristics of an alternative cladding material may be represented by the modeling practices used in an analysis performed with the FVI SBLOCA methodology. In the event limited data is available for performing an assessment, the licensee's response states that compensatory actions may be taken, including defining a conservative peak cladding temperature penalty that would be intended to account for data limitations.

In accordance with Criterion III of Appendix B to 10 CFR 50, the establishment of design control measures is required to assure that applicable regulatory requirements and design basis information is correctly translated into specifications, drawings, procedures, and instructions. Criterion III identifies that such design control measures shall be applied to, among other things, accident analyses. Criterion III further requires that measures shall also be established for the *selection and review for suitability of application* of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components. To assure that analytical predictions made by the proposed FVI SBLOCA methodology would representatively model the actual plant design, in accordance with the above regulatory requirements, the following information is needed to support the licensee's proposed fuel modeling practices:

(a) Please identify the document that contains the established process for evaluating alternative cladding materials and summarize the key elements of this process.

(b) Please clarify whether the established process applies solely to the evaluation of alternative cladding materials or whether it applies more generally to any fuel properties which may diverge from those that have been explicitly evaluated using the FVI SBLOCA methodology. If the established process applies only to alternative fuel cladding materials, then please further discuss how impacts of differences in other fuel properties would be assessed.

(c) Please adequately describe and justify in particular the process for defining a conservative penalty on peak cladding temperature under conditions where available data is limited, considering that data limitations may challenge the capability to define a conservative penalty.

RAI 16 S1

The response to RAI 16 provided evidence generated during the review of a design certification for an evolutionary pressurized water reactor design, suggesting that **[[** [redacted] **]]**. Considering the following:

- Whereas the limiting SBLOCA event for the design certification application exhibited **[[** [redacted] **]**;
- The RAI response associated with the design certification review (ML090970699) indicated that the time required for **[[** [redacted] **]]**;
- For some of the limiting **[[** [redacted] **]]** described in ANP-3467P and ANP-3676P, the **[** [redacted] and
- For Surry in particular, the **[[** [redacted] **]]** during the transient,

The NRC staff is not able to conclude that the effects of TCD **[[** [redacted] **]]**. Paragraph 10 CFR 50.46(a)(1)(i) requires the consideration of postulated LOCAs of sufficient sizes, locations, and other properties sufficient to provide assurance that the most severe hypothetical LOCAs are calculated. In addition, Paragraph 1, "The initial stored energy in the fuel," of Section A, "Sources of heat during the LOCA," of Part I, "Required and Acceptable Features of the Evaluation Models," of Appendix K, "ECCS Evaluation Models," to 10 CFR 50 requires that the UO_2 thermal conductivity be evaluated as a function of burnup and temperature. **[** [redacted] **]**. Provide sufficient information that is specifically applicable to the requesting plants to demonstrate that the existing approach **[** [redacted] **]**, consistent with the required and acceptable features of ECCS evaluation models set forth in Appendix K to 10 CFR 50. If the existing approach **[[** [redacted]

RAI 18-S1: ADEQUACY OF RUPTURE STRAIN VS. RUPTURE TEMPERATURE CORRELATION

Regulatory Basis

Appendix K to 10 CFR 50, Part I, "Required and Acceptable Features of the Evaluation Models," Section B, "Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters," requires, in part, that, "To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated."

In addition to the Appendix K requirement, 10 CFR 50.46(a)(1)(i) requires the calculation of a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe hypothetical loss-of-coolant accidents are calculated. Paragraph 10 CFR 50.46(b)(1) requires that the peak cladding temperature remain below 2200 °F, and 10 CFR 50.46(b)(2) limits the maximum amount of local oxidation to 17% of the cladding thickness before oxidation, with oxidation calculated for both cladding surfaces if rupture is calculated to occur.

Description of Rupture Strain/Temperature Relationship

The response to RAI 18 **[**[redacted]**]]**

> **[[**[redacted]**]]**

The NRC staff relied on this general description of the high-temperature plastic deformation behavior of zirconium alloys, in concert with **[[**[redacted]**]]**.

Review of RAI 18 Response

[[[redacted]**]**

[[[redacted]]].

Assessment of RAI 18 Response Against North Anna Rupture Results

The response to RAI 18 states, **[[** [redacted] **]]**

The North Anna break spectrum analysis described in ANP-3467P included **[[** [redacted] **]]**

Consideration of an Alternative, Advanced Zirconium Alloy Cladding

Consider the following discussion, as provided by Framatome Cogema Fuels (now Framatome) in Appendix C to BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," in **[** [redacted] **]]**:

> **[[** [redacted] **]**
>
> **[[** [redacted]

[redacted]]

Additional consideration of M5 data identifies [[redacted]]]

Supplemental Request

Given that the FVI SBLOCA analyses **[[** [redacted] **]]**, additional justification is required **[[** [redacted] **]]** remains consistent with 10 CFR 50.46(a)(1)(i) and 10 CFR 50, Appendix K requirements identified above. Please provide an adequate demonstration that **[[** [redacted] **]]** does not underestimate the degree of swelling, consistent with the applicable Appendix K requirements identified above, and ensure that the demonstration is applicable within the range of peak fuel cladding temperatures permitted by the acceptance criteria contained in 10 CFR 50.46(b)(1), i.e., up to 2200 °F. Provide justification that the model adequately considers **[[** [redacted] **]]**, regarding overall effect on essential figures of merit.

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