NRR-DMPSPEm Resource

From:	Hon, Andrew
Sent:	Thursday, May 9, 2019 12:03 PM
To:	Yodersmith, Stephen B
Cc:	Galvin, Dennis
Subject:	Brunswick Unit 1 and Unit 2 Request for Additional Information related Transition to Framatome ATRIUM-11 Fuel (EPID: L-2018-LLA-0273)
Attachments:	Brunswick A11 RAIs_Redacted.pdf

By letter dated October 11, 2018, Duke Energy submitted a license amendment request for Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick) to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM-11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain (Agencywide Documents Access and Management System (ADAMS) Accession No. ML1828A395).

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is transmitted via protected method because proprietary information is included. The proposed questions were discussed by telephone with your staff on April 26, 2019. Your staff confirmed that the request for additional information (RAI) was understood, the proprietary information marking as redacted in the attached was correct, and agreed to provide a response by June 18, 2019.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. Please note that if you do not respond to this request by the agreed-upon date or provide an acceptable alternate date, we may deny your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Section 2.108. If circumstances result in the need to revise the agreed upon response date, please contact me.

Andy Hon, *PE*
Project Manager (Brunswick Nuclear Plant 1 & 2, Sequoyah Nuclear Plant 1 & 2)
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
301-415-8480
OWFN O8H19
Mail Stop O8G-9a
andrew.hon@nrc.gov

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Recipients:
"Galvin, Dennis" <Dennis.Galvin@nrc.gov>
Tracking Status: None
"Yodersmith, Stephen B" <Stephen.Yodersmith@duke-energy.com>
Tracking Status: None

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REQUESTS FOR INFORMATION FOR

BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2

TO SUPPORT REVIEW OF THE LICENSE AMENDMENT REQUEST

REGARDING APPLICATION OF FRAMATOME METHODOLOGIES

TO SUPPORT TRANSITION TO ATRIUM-11 FUEL

Facility	Docket	EPID
Brunswick Steam Electric Plant, Unit 1	50-325	L-2018-LLA-0273
Brunswick Steam Electric Plant, Unit 2	50-324	

By letter dated October 11, 2018, Duke Energy submitted a license amendment request for Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick) to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM-11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain (Agencywide Documents Access and Management System (ADAMS) Accession No. ML1828A395). Upon review of the submittal, the staff has determined the following additional information is necessary to continue the review.

The proprietary information in this document is marked with double brackets and bold font such as **[[Example]]**.

The Regulatory Basis

- Title 10 Code of Federal Regulation Chapter (10 CFR) 50.46 *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*,
- 10 CFR 50.62 *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light- water-cooled nuclear power plants*, and
- 10 CFR 50 Appendix A General Design Criteria (GDC):
    - 10 *Reactor Design*
    - 12 *Suppression of Reactor Power Oscillation*s,
    - 13 *Instrumentation and Control*
    - 15 *Reactor Coolant System Design*
    - 16 *Containment Design*
    - 20 *Protection System Functions*
    - 25 *Protection System Requirements for Reactivity Control Malfunctions*, and

    - 26 *Reactivity Control System Redundancy and Capability*
    - 38 *Containment Heat Removal*, and
    - 50 *Containment Design Basis*

- 10 CFR 50 Appendix K *Emergency Core Cooling System, Evaluation Models*

Specific Request for Additional Information

Anticipated Transient without Scram with Instability (ATWS-I)

1. Justify the use of a feedwater temperature reduction rate of 1.3°F/s, as well as the initial delay time (if any) before feedwater temperature reduction begins in the ATWS-I analysis.

2. Discuss how the gap conductance sensitivity will be addressed when fuel design changes occur and provide results for ATRIUM-11 fuel.

3. Confirm that the steam line and valve modeling options accurately capture the expected plant-specific system performance during ATWS-I events.

4. Provide a justification that the ATWS-I analyses based on the reference core will bound all expected future core designs. As part of this discussion, address transition cores and core design specific considerations that may affect local stability characteristics, such as nodal variations, control rod patterns, and operating strategies.

5. Two events are considered potentially limiting in the ATWS-I transient scenario: two reactor pump trip (2RPT) and turbine trip with bypass (TTWB). Brunswick analyzed the TTWB event and, since instability and dryout/rewet occurred, the 2RPT event was unanalyzed per the Calculational Procedure in Section 8.0 of the submittal. In order to assure the limiting event was analyzed, provide results for the 2RPT ATWS-I event, with justification given for the operator action time assumptions used. Certain changes in plant design or operation may affect stability behavior for these events. Discuss how the TTWB event will be confirmed to remain limiting relative to the 2RPT event if changes are made to the plant design or operation that may affect stability behavior during anticipated transient without scram (ATWS), such as: turbine bypass capability, fraction of steam-driven feedwater pumps, and changes expected to increase core inlet subcooling during ATWS events.

6. Justify that the selected settings and modeling options are appropriate, including core and vessel nodalization, time step control parameters, and noise parameters. Discuss how the modeling is consistent with the characteristics of Brunswick and the validation basis for the proposed RAMONA5-FA ATWS-I methodology.

7. Understanding that both **[[**

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8. Tables 7-1 and F-1 of ANP-3694P indicate a trend of **[[**

]]. Provide an explanation for this observed trend.

<u>Anticipated Operating Event (AOO) and ATWS</u>

9. ANP-3702P provides a subset of the events analyzed in the Brunswick Chapter 15 Updated Final Safety Analysis Report (UFSAR) and covered by the AURORA-B AOO/ATWS methodology. To ensure the methodology is implemented appropriately for the events not covered in ANP-3702P and to ensure that the analysis of these events is sufficient to meet GDCs 10, 13, 15, 20, 25, and 26 and ATWS acceptance criteria, provide the following:

    a. Describe how each Chapter 15 UFSAR event (that is covered by the AUORAB-AOO/ATWS methodology) will be analyzed in the AURORA-B AOO methodology framework (e.g., a table identifying FSAR Section/Event Name/Disposition)

    b. Describe how the methodology is implemented (including steps prior to the execution of the uncertainty analysis) to ensure there is appropriate coverage of operational power/flow statepoints, equipment-out-of-service conditions, time-in-cycle, etc.

10. To ensure there is appropriate coverage of the parameters used in the uncertainty analysis and to ensure there is no significant trends with respect to the uncertainty parameters in the results such that the Brunswick implementation of the AURORA-B methodology is sufficient to meet GDCs 10, 13, 15, 20, 25, and 26 and ATWS acceptance criteria, provide the following for the load rejection no bypass (LRNB) event at 100% power / 104.5% flow and main steam isolation valve (MSIV) closure ATWS event at 100% power and 85% flow:

    a. The sampled values of the uncertainty parameters for all cases executed in the set

    b. The figure of merit (FoM) results for all cases executed in the set

<u>Loss-of-Coolant Accident (LOCA)</u>

11. Please justify the statement on page 1-2 of ANP-3674P that “the limiting break will not change with exposure or nuclear fuel design.” Although the NRC staff’s safety evaluation on ANP-10332P found that the AURORA-B LOCA evaluation model may **[[**

    ]. Hence, the requested information is necessary to justify satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that a sufficient number of postulated scenarios has been considered to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

12. Please justify the conclusion in ANP-3674P that postulated breaks in the reactor water cleanup system and instrument lines **[[**

].

Hence, the requested information is necessary to justify satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that a sufficient number of postulated scenarios has been considered to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

13. Although figures of merit are provided for limiting scenarios, ANP-3674P does not demonstrate that the AURORA-B LOCA evaluation model **[[**

]]. The requested information is necessary to justify satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that a sufficient number of postulated scenarios has been considered to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

14. The predicted peak cladding temperature for the SF-BATT single failure **[**

]]. The requested information is necessary to justify a conservative calculation of the figures of merit in demonstrating satisfaction of the relevant acceptance criteria specified in 10 CFR 50.46(b).

15. Table 9.1 of ANP-3674P shows that the calculated maximum local cladding oxidation **[[**

]. The requested information is necessary to justify satisfaction of the acceptance criterion for local cladding oxidation (i.e., total oxidation) specified in 10 CFR 50.46(b)(2).

16. Limitation and Condition 15 from the NRC staff's draft safety evaluation on ANP-10332P requested that licensees **[**

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ANP-3674P does not provide sufficient information to **[[**

]. The requested information is necessary to justify satisfaction of the relevant acceptance criteria specified in 10 CFR 50.46(b).

17. Please demonstrate that either (1) the calculations performed for Brunswick either directly satisfy Limitation and Condition 19 from the NRC staff's draft safety evaluation on ANP-10332P or (2) the predicted results for Brunswick presented in ANP-3674P would provide more conservative results than simulations that directly satisfy Limitation and Condition 19. The requested information is necessary to justify satisfaction of paragraph I.C.4.e of Appendix K to 10 CFR 50.

18. Please identify each operating domain for which Brunswick is currently licensed (e.g., expanded operating domains on power-to-flow map, equipment-out-of-service conditions). Please clarify whether each licensed operating domain has been analyzed explicitly for the LOCA event, or whether it has been dispositioned based upon qualitative factors. For each licensed operating domain, provide the analytical results or qualitative rationale demonstrating the condition is non-limiting. The requested information is necessary to justify satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that a sufficient number of postulated scenarios has been considered to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

19. As described in ANP-3674P, the LOCA analysis performed for Brunswick considered a **[[**

]. The requested information is necessary to justify satisfaction of the requirement in 10 CFR 50.46(a)(1)(i) that a sufficient number of postulated scenarios has been considered to provide assurance that the most severe postulated loss-of-coolant accident has been calculated.

20. Please show that Limitation and Condition 35 from the NRC staff's draft safety evaluation on ANP-10332P has been satisfied by providing **[[**

]. The requested information is necessary to justify satisfaction of the relevant acceptance criteria specified in 10 CFR 50.46(b) and the requirements of Paragraph II.3 of Appendix K to 10 CFR 50.

21. Please show that Limitation and Condition 37 from the NRC staff's draft safety evaluation on ANP-10332P has been satisfied by providing **[[**

]]. The requested information is necessary to justify satisfaction of the relevant acceptance criteria specified in 10 CFR 50.46(b) and the requirements of Paragraph II.3 of Appendix K to 10 CFR 50.

22. The NRC staff's review of the results presented in ANP-3674P observed notable differences relative to **[[**

]].

Please clarify the basis for these observed differences. The requested information is necessary to justify satisfaction of the relevant acceptance criteria specified in 10 CFR 50.46(b).

Best Estimate Enhanced Option-III (BEO-III) with Confirmation Density Algorithm (CDA)

23. For cycle operation that differs significantly from the original cycle design, describe and justify the process for evaluating whether the analysis continues to bound actual plant operation or whether additional analysis is necessary.

24. Provide sensitivity studies on timestep size and vessel nodalization to demonstrate that potential perturbations to discretization would not have an undue impact on calculated figures of merit or change the sensitivities to statistical parameters.

25. Provide the following clarifications for the **[[ ]]** BEO-III analysis:
    a. How is the duration of the **[[ ]]**
    b. What method is used to calculate the **[[ ]]** from the RAMONA5-FA time-dependent results?
    c. How is the perturbation amplitude **[[ ]]** determined?
    d. Explain why the **[[ ]]**.
    e. Provide plot of core pressure drop **[[ ]]**

26. Address the impact of including medium-importance phenomena in the calculation of relevant figures of merit. Perform an updated BEO-III statistical analysis including all medium-importance phenomena, or justify the exclusion of any particular medium-importance phenomena.

27. The capability to characterize the oscillation mode (i.e., as in-phase or out-of-phase) for a predicted instability condition may provide information that is relevant to (1) understanding the physical behavior of the reactor, (2) assessing code predictions against a priori expectation, and (3) assuring that uncertainty characterizations are appropriate.
    a. Please clarify whether the plant-specific BEO-III methodology is capable of characterizing the predicted oscillation mode. If such capability exists, please describe and justify the approach used to perform this characterization.
    b. Please clarify whether any parameters and phenomena considered in the uncertainty analysis (including any medium-importance phenomena incorporated to address RAI 26) may have a significantly different impact on predicted figures of merit, depending on whether the oscillation mode for the cases analyzed tends to be in-phase or out-of-phase.
    c. Provide an assessment of which oscillation modes were observed in the BEO-III analysis for Brunswick. Was significant variation observed among the statistical cases, in terms of which oscillation mode was dominant? Did observable trends

exist in which particular parameter values tended to result in particular mode(s) becoming dominant?

d. Did the most limiting 95/95 FOM cases tend to be associated with a particular oscillation mode?

27. Perform plant-specific BEO-III calculations for the 100%-power / 85%-flow single-recirculation-pump-trip event and provide results including the calculated values for oscillation period, to ensure that all anticipated oscillations remain within the period detection bounds for the Brunswick CDA implementation.

28. Particular statistical cases were **[[**

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29. Justify the dispositioning of the following phenomena as “Low Importance” in the BEO-III phenomena identification and ranking table (PIRT), given their potential significance for stability and, in some cases, their inclusion in the AURORA-B AOO (ANP-10300P, Rev 1) statistical sampling for non-pressurization transients:
    - **[[**
    -
    - **]]**

30. In Stage 3 of the multistage analysis, **[[**

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<u>Containment</u>

31. In section 7.3 of ANP-3705P, the licensee states that fuel design differences may impact the power and pressure excursion experienced during an ATWS event. The licensee further states that ATRIUM-10XM analysis bounds the ATRIUM-11 fuel because **[[**

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a. Describe the analysis done to justify that **[**

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b. Provide quantitative results for the containment pressure and suppression pool temperature response changes due to the change in fuel type. Describe the analyses performed to confirm the ATRIUM-10XM analysis bounds the ATRIUM-11 fuel transition.

c. Containment heatup is directly impacted by the stored energy within the fuel and the decay heat. Provide a quantitative comparison of the decay heat between the ATRIUM-10XM and ATRIUM-11 fuel.

32. No additional events were listed in the application as having had an impact from the transition to ATRIUM-11 fuel. Explain any changes that were made to any analyses which impact the mass and energy release during an accident or a special event (station blackout or fire event).