TVA TENNESSEE VALLEY AUTHORITY

1101 Market Street, Chattanooga, Tennessee 37402

CNL-22-066

July 18, 2022

10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) - Supplement 3, Response to Request for Additional Information (EPID L 2021-LLA-0132)**

References:

1. TVA letter to NRC, CNL-21-053, "Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535)," dated July 23, 2021 (ML21204A128 and ML21204A129)

2. NRC Electronic Mail to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information re LAR to Use Advanced Framatome Methodologies in Support of ATRIUM 11 Fuel (EPID L-2021-LLA-0132)," dated June 3, 2022 (ML22160A474 and ML22160A681)

In Reference 1, Tennessee Valley Authority (TVA) submitted a request for a Technical Specification (TS) amendment for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. The license amendment request (LAR) revises TS 5.6.5.b, "Core Operating Limits Report (COLR)," to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM™[1] 11. Additionally, the LAR requests adoption of Technical Specification Task Force (TSTF)-564-A, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the BFN TS. The proposed amendment revises the TS safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for an SL.

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In Reference 2, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) and requested that TVA respond by July 18, 2022. Enclosure 1 to this letter provides the TVA response to the RAI.

Enclosure 1 to this letter contains information that Framatome, Inc. (Framatome) considers to be proprietary pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). Enclosure 2 to this letter provides a non-proprietary version of the information provided in Enclosure 1. Enclosure 3 provides the Framatome affidavit supporting this proprietary withholding request. Therefore, TVA requests that Enclosure 1, which is proprietary to Framatome, be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of the item listed above or the supporting Framatome affidavit should reference the corresponding report and should be addressed to Alan Meginnis, Framatome, Manager, Product Licensing, 2101 Horn Rapids Road, Richland, WA 99354.

This letter does not change the no significant hazards considerations or the environmental considerations contained in Reference 1. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the non-proprietary enclosures to the Alabama Department of Public Health.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Stuart L. Rymer, Senior Manager, Fleet Licensing, at slrymer@tva.gov.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 18th day of July 2022.

Respectfully,

James Barstow

Digitally signed by Rearden, Pamela S
Date: 2022.07.18 17:02:50 -04'00'

James Barstow
Vice President, Nuclear Regulatory Affairs & Support Services

Enclosures:

1. Response to NRC Request for Additional Information (Proprietary version)
2. Response to NRC Request for Additional Information (Non-proprietary version)
3. Framatome Affidavit

cc: (Enclosures):

NRC Regional Administrator – Region II
NRC Project Manager – Browns Ferry Nuclear Plant
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama Department of Public Health (w/o Enclosure 1)

Enclosure 1

Response to NRC Request for Additional Information
(Proprietary version)

Enclosure 2

Response to NRC Request for Additional Information
(Non Proprietary)

Browns Ferry Advanced Methods License Amendment Request – Response to Request for Additional Information

ANP-4006NP
Revision 1

July 2022

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	p. 2-1	Last line of the page, provided the definition of ECPR as experimental critical power ratio
2	p. 2-6	Updated the Framatome response
3	p. 2-7	Added additional text within the proprietary brackets Added a pointer to newly added Reference 11 for AN-NF-82-06(P)(A)
4	p. 2-10	Updated text in the second paragraph of the Framatome response
5	p. 2-16	Added text at the beginning of the Framatome response
6	p. 2-22 and p. 2-23	Added text at the end of the first sentence of the Framatome response Removed Proprietary marking in the NRC request of part 10(a) and 10(b)
7	p. 2-25	Updated the first and second sentence of the Framatome response
8	References	Updated Reference 1 Added Reference 11

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1.0 INTRODUCTION

By letter dated July 23, 2021, the Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) to the U. S. Nuclear Regulatory Commission (NRC) for the Browns Ferry Nuclear Plant Units 1, 2, and 3 (Browns Ferry). The amendment would revise the Browns Ferry Technical Specification 5.6.5.b, Core Operating Limits Report (COLR), to allow the application of advanced Framatome Inc., methodologies for determining the core operating limits in support of the loading of the Framatome, Inc. ATRIUM 11 fuel type into the Browns Ferry cores. Upon review of the submittal, the NRC staff provided requests for additional information (RAI) in a letter dated June 3, 2022 (Reference 1). This report provides responses to these RAIs.

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

2.0 SNSB REGULATORY BASES AND RAIs

SNSB RAI 1:

Regulatory Basis:

Atomic Energy Commission (AEC) CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A) states:

> The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Request:

In ANP-3857P/NP, Revision 2, “Design Limits for Framatome Critical Power Correlations,” Table 1, for the use of the critical power correlation in topical report ANP-10335P-A* for ATRIUM 11 fuel, ANP-10335P-A contains limitations and conditions in Section 4.0 of the NRC’s safety evaluation. However, it is not apparent that Framatome has addressed the L&Cs for this application. Provide a disposition for each L&C.

Response:

The use of the design limits from ANP-3857P/NP must consider the broader context associated with their use: TSTF-564, Rev. 2 (ADAMS Accession No. ML18297A361), which has been approved by the NRC. A new $MCPR_{95/95}$ limit is described which is determined from the critical power correlation experimental critical power ratio (ECPR)

* Framatome, Inc., Topical Report ANP-10335P-A, Revision 0, *ACE/ATRIUM 11 Critical Power Correlation*, May 2018 (ADAMS Package Accession No. ML18207A382).

and ECPR standard deviation. This value does not consider most uncertainties that affect the MCPR operating limit (OLMCPR).

In TSTF-564, Rev. 2, Section 3, last paragraph states:

> *"The LCO 3.2.2 limits (i.e., the OLMCPR values) are not changed and will be based on the existing SLMCPR, referred to as $MCPR_{99.9\%}$. The OLMCPR will continue to be determined based on the transient ΔCPR components and the cycle-specific $MCPR_{99.9\%}$ value that will be included in the COLR. Therefore, the margin to boiling transition remains unchanged."*

The $MCPR_{99.9\%}$ limit is calculated using previously approved methodologies. It accounts for all significant cycle-to-cycle, fuel, and plant uncertainties.

The ACE/ATRIUM 11 limitations and conditions are fully accounted in the determination of the LCO 3.2.2 limits. The disposition of these limitations and conditions within LCO 3.2.2 is provided in Table 2.1.

Table 2-1 Limitations from ACE/ATRIUM 11 Critical Power Correlation Topical Report

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
1	The ACE/ATRIUM 11 correlation shall not be applied outside of the parameter ranges presented in Table 2.1 of ANP-10335P. Because the testing did not include flow in the internal water canister, the limits on mass flow rate are imposed on the mass flow rate in the heated section of the bundle (i.e., they do not include bypass flow that would be included if the bundle inlet mass flow rate were to be used). Also note that while Framatome did not specify [[ ]]. Additionally, the LPF limit of [[ ]] can be exceeded only for perturbed conditions in MCPR safety limit Monte Carlo calculations and for bundles that can be shown to be non-limiting (e.g. ., high burnup or controlled bundles).	The ACE correlation is implemented in the code library ACELIB. ACELIB is used by all codes that require determination of critical power for ATRIUM 11. Limitations on parameter ranges of mass flow rate, inlet enthalpy, and pressure are enforced within ACELIB. [[ ]]

Table 2-1 Limitations from ACE/ATRIUM 11 Critical Power Correlation Topical Report (continued)

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
2	For bundles with LPFs greater than [[ ]]. The following increased [[ ]] uncertainties shall be applied to the following listed rod positions [[ ]]: [ ]	The higher uncertainty associated with peaking factors greater than [[ ]] is applied within SAFLIM3D (Reference 3) which is used to determine the Minimum Critical Power Ratio Safety Limit (SLMCPR) also referred to as $MCPR_{99.9\%}$. The rod position dependent [[ ]] shown in the Limitations and Conditions are also applied in SAFLIM3D. These are [[ ]] .
3	Application of the ACE/ATRIUM 11 correlation in a transient analysis methodology requires verification that the correlation conservatively predicts CP compared to test data and demonstrates similar behavior compared to other implementations of the correlation. Framatome shall not apply the ACE/ATRIUM 11 correlation in transient analysis methodologies other than XCOBRA-T and AURORA-B without first verifying the appropriate correlation behavior and conservatism.	The transient analysis methodology applied to the ATRIUM 11 analyses is AURORA-B (Reference 4). The verification of applicability of the ACE/ATRIUM 11 correlation with this methodology was demonstrated in Reference 2 (ANP-10335Q2P, Section 2.0).

SNSB RAI 2:

Regulatory Basis:

Same as in SNSB RAI 1.

Request:

ANP-3859, Section 3.2 states that **[[**

]] However, no explanation is provided to support this statement. Explain why the **[[**
]] in the above determination.

Response:

The radial distribution will have an impact on overall flow distribution. If radial power in the hot channels were to be increased, the increased voiding would increase the two-phase pressure drop in those channels. However, in order to maintain the core average power, the power in other channels will need to be decreased which will lead to a decrease in two-phase pressure drop in those channels. All channels communicate to a common channel inlet as well as a common channel outlet which forces all pressure drops in the core to be equal. In order to maintain that equal pressure drop the flow in the hot channels will be reduced while the flow in the cooler assemblies will be increased. While changes in radial power distributions will have some impact on core pressure drop, the core flow redistribution will mean that any impacts are likely to be small. The key to the hydraulic compatibility analysis is to ensure that each fuel design evaluated has the same basis, therefore each fuel design must be evaluated at the same radial power to provide relative comparisons.

SNSB RAI 3

Regulatory Basis:

Same as in SNSB RAI 1

Request:

In ANP-3905, Section 7.2, Framatome states that for single loop operation (SLO) a 0.85 multiplier is applied to the two-loop maximum average planar linear heat generation rate (MAPLHGR) limit resulting in an SLO MAPLHGR limit of **[[ ]]** kW/ft. However, no explanation is provided for the selection of this multiplier. Explain how the multiplier 0.85 was selected to determine the maximum MAPLHGR for SLO.

Response:

The Framatome approach for performing SLO LOCA calculations is to require that the two-loop operation (TLO) PCT is always higher than the SLO PCT. This is accomplished by applying a multiplier on the TLO MAPLHGR limit to determine a reduced MAPLHGR limit to be used for SLO. For LOCA analyses with ATRIUM 11 fuel, it has been determined that a 0.85 multiplier is adequate to ensure the limiting TLO PCT bounds the limiting SLO PCT. If a future break spectrum evaluation determined that the 0.85 multiplier was not adequate to make TLO PCT bounding, a smaller multiplier would be selected which would make TLO PCT limiting.

SNSB RAI 4:

Regulatory Basis:

Same as in SNSB RAI 1

Request:

In ANP-3905, Appendix A, limitation and condition 11 states:

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[** **]]**

The vendor's disposition is as follows:

> BWR [Boiling Water Reactor] fuel rods are **[[** **]].**

Provide the basis for **[[** **]].**

Response:

The basis for **[[** **]]** is based on Figure 2-1 which is from the approved rupture model in XN-NF-82-07(P)(A), Reference 11, and used by the AURORA-B LOCA method as indicated in ANP-3905P Section 4.1.

Figure 2-1 S-RELAP5 BWR Burst Strain Model

SNSB RAI 5:

Regulatory Basis:

10 CFR 50.46(b)(1), Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

Request:

ANP-3905, Table 4.5 states the low pressure coolant injection (LPCI) injection valve stroke time to be 40 seconds.

Referring to ANP-3905, Table 5.1, the single failure (SF) cases SF-BATT [SF-battery], SF-DGEN [SF-diesel generator], SF-HPCI [SF-high pressure coolant injection], and SF-ADS [SF-automatic depressurization system] use either one LPCI (two pumps) loop or two LPCI (four pumps) loops.

During normal operation, in the scenario in which the residual heat removal (RHR) system is placed in the suppression pool cooling mode or flow test mode, the RHR system test line isolation valve through which water returns to the suppression pool is open. The Browns Ferry Updated Final Safety Analysis Report (UFSAR), Amendment 29, section 7.4.3.5.4 states the automatic closing time for this valve for LPCI operation is 90 seconds.

In the analysis based on the single failures noted above, for a loss-of-coolant (LOCA) in the scenario while the RHR system is in the suppression pool cooling mode during normal operation, with the return flow through the test line isolation valve, the unit is placed in Limiting Condition of Operation (LCO) 3.6.2.1. During the period in the which the unit is in LCO, the design basis single failure assumption is temporarily relaxed. However, in the LPCI flow test mode (Surveillance Requirement 3.5.1.6), there is no LCO associated with this mode and, therefore, the design basis does not allow a single failure while operating in this mode.

While RHR is operating in the LPCI flow test mode, the test line isolation valve should automatically close on receiving a LOCA signal in 90 seconds, while the LPCI injection valve fully opens in 40 seconds from the same signal. During the 50 seconds time difference (between the closing time of test line isolation valve and the opening time of

the LPCI injection valve) some of the LPCI flow will bypass to the suppression pool and, therefore, the reactor will not receive the fully rated LPCI flow.

In the analysis based on the single failures noted above, for a LOCA in the scenario while the RHR system is in the test mode during normal operation, with the test line isolation valve partially open for 50 seconds, provide the following:

(a) Confirm that partially closed (instead of fully closed) test line isolation valve was considered by not crediting the fully rated LPCI flow. Provide the LPCI flow rate credited in the first 90 seconds from the LOCA signal and the fully rated LPCI flow credited after 90 seconds from LOCA signal.

Response:

The scenario postulated in SNSB RAI 5 refers to the LPCI valve stroke times in BFN UFSAR Section 7.4.3.5.4 to establish that there could be up to 50 seconds of LPCI flow diversion to the suppression pool during a LOCA with Loss of Offsite Power (LOOP). This stated duration of LPCI flow diversion is based on a comparison of the closing stroke time of one particular valve in the LPCI test return line, and the opening stroke time of the LPCI injection valve. The LOCA analysis presented in ANP-3905P Revision 1 does not account for the scenario of LPCI flow being partially diverted thru the test line back to the suppression pool.

The Residual Heat Removal System test line isolation valve mentioned in the BFN UFSAR Section 7.4.3.5.4 has a closure time of 90 seconds. However, there is another valve downstream of this valve that is partially closed during the test, and is the valve used to throttle the flow from the LPCI pump during the testing. This valve is either FCV-74-59 or FCV-74-73, depending on which pair of LPCI pumps are being tested (see BFN UFSAR Figure 7.4-6a Sheet 1). During testing, this valve is positioned such that it will stroke from the test position to fully closed in a maximum time of 49 seconds (significantly shorter than the 90 second closing stroke of the upstream valve mentioned in the RAI). It is the FCV-74-59 or FCV-74-73 valve that determines the time at which the affected return line is fully isolated on a LOCA signal. These valves have an associated surveillance test procedure that ensure the 49 second closing time criterion is met.

The actual duration of any potential LPCI flow diversion is more complex than just comparing the relative stroke times of the LPCI injection valves (FCV-74-53 and FCV-74-67) and the test valve in the return line which is throttling the pump flow. The LPCI injection valves and the test valves (FCV-74-59 or FCV-74-73) all regain electrical power in a LOCA/LOOP at the same time (via the emergency buses powered by the diesel generators). The test valves in the return line would begin to stroke closed once the valve obtains power. However, the LPCI injection valves will not start to stroke open until the reactor pressure permissive setpoint is also cleared. For this reason, the extent and duration of LPCI flow diversion thru the test line is also a function of the size and location of the break in the recirculation line, as those factors (along with ADS) influence when the injection valve pressure permissive is satisfied.

For this reason, the LPCI flow as a function of time will also vary depending on the specifics of the break size and location. As noted below in the response to part (b), the limiting break and single failure combinations for Browns Ferry do not credit any LPCI injection at all. This is the reason why Figure 6.7 of ANP-3905P Revision1 shows zero LPCI flow for the entire event duration.

(b) If the fully rated constant LPCI flow is used in the analyses starting at 40 seconds from LOCA initiation, justify.

Response:

As noted in the response to part (a), consideration of short term LPCI flow diversion in a LOCA/LOOP which initiates during testing of a LPCI pump is not considered in the analysis presented in ANP-3905P Revision 1. The Framatome LOCA methods do model and credit LPCI flow thru the LPCI injection valves as they are stroking open, once the LPCI pumps are at rated speed. Rated LPCI flow in the ANP-3905P Revision 1 analyses only occurs when the injection valves are fully open 40 seconds after the injection valves start to stroke open, with the LPCI pumps at rated speed.

Given that the scenario postulated in the RAI is not explicitly accounted for in the LOCA analyses, the following sections provide both qualitative and quantitative discussions of the impacts of LPCI flow diversion thru the test return line. The discussion will differentiate between breaks on the discharge side of a

recirculation loop, and breaks occurring on the suction side of a recirculation loop.

The scenario is conservatively bounded by not crediting any LPCI flow for the limiting case. Table 6.2 in ANP-3905P Revision 1 shows the limiting and near limiting breaks occur in the pump discharge line for single failures SF-BATT|BB and SF-BATT|BA. Those pump discharge cases have no LPCI as shown by the ECCS availability in Table 5.1 of ANP-3905P Revision 1, so they are not affected by delayed LPCI flow.

LPCI is credited for pump suction breaks so the scenario would delay LPCI flow for those cases. Bounding sensitivity calculations were performed using a 49 second LPCI valve opening time and no credit for LPCI flow until the valve is fully opened. This delays all LPCI injection until after the test line isolation valve closes and the LPCI valve opens. The pump suction cases with the highest PCTs are 1.0 DEG breaks, which depressurize below the LPCI pressure permissive before power is available to the valves and maximize the potential impact of a LPCI delay. Sensitivity calculation results for the 1.0 DEG pump suction breaks at all state points, both axials and the two failures (SF-BATT|BA and SF-BATT|BB) in Table 2-2 show the PCTs remain substantially non-limiting and do not affect the limiting PCT or oxidation results reported in ANP-3905P Revision 1.

Table 2-2 Sensitivity PCTs for 1.0 DEG Pump Suction Breaks

SNSB RAI 6:

Regulatory Basis:

Same as in SNSB RAI 1

Request:

ANP-3904, Table 3.1, “Disposition of Events Summary for Introduction of ATRIUM 11 Fuel at Browns Ferry,” lists two events which state the events are expected to be non-limiting. The events are UFSAR, section 14.5.2.5, “Turbine bypass valves failure following turbine trip (TTNB), high power,” and UFSAR, section 14.5.2.6, “Turbine bypass valves failure following turbine trip (TTNB), low power.” The disposition status of these events is described as “Address initial reload” and “No further analysis required” respectively and are stated to be *generally bounded* [emphasis added] by the FSAR Section 14.5.2.2 event. However, no information is provided explaining how these events are verified to be non-limiting for each reload, or justifying why such a verification is not necessary. If the events do not prove to be non-limiting, explain the process to ensure protection for each reload.

Response:

The objective of the disposition of events is to identify the limiting events which need to be analyzed to support plant operation. As discussed in ANP-3904P Section 3.2, a cycle specific calculation plan is developed to identify the analyses to be performed as part of the licensing campaign. The calculation plan is based on the results of the disposition of events. All events that are not dispositioned as “no further analysis required” are addressed in the calculation plan. An event for which the disposition status is “address for the initial reload” with the comment that it is expected to be bound by another event, will be addressed in the calculation plan. If the event has been shown to be non-limiting based on a previous cycle analysis, the calculation plan will identify the licensing campaign in which the analysis was performed and state that no further analysis is needed for the upcoming cycle. If the initial or subsequent analysis does not conclude the event is non-limiting, the calculation plan will identify the event as needing analysis for the upcoming cycle.

It is noted that ANP-3904P is a demonstration of the applicability of the AURORA-B methodology to Browns Ferry for transient events that are typically limiting and does not represent licensing analysis results for any particular cycle.

SNSB RAI 7:

Regulatory Basis:

Same as in SNSB RAI 1

Request:

ANP-3904, section 4.1.4 provides the American Society for Mechanical Engineers (ASME) maximum overpressurization analyses based on UFSAR Section 14.5.2.7 MSIV closure event. As stated in ANP-3904, Table 3.1, in the 'Comments' column, this event is bounded by the FSAR Section 14.5.2.2.4 LNRB with EOC-RPT-OOS [generator load rejection no bypass with end of cycle recirculation pump trip out of service] event which is a potentially limiting abnormal operational transient (AOT). Provide reason(s) for not performing the overpressurization analysis based on the more limiting UFSAR 14.5.2.2.4 AOT event.

Response:

The NRC RAI is asking TVA to provide reason(s) for not performing an overpressurization analysis of the UFSAR 14.5.2.2.4 event. TVA notes that the referenced UFSAR Section refers to MCPR transients and not overpressurization events.

Table 3.1 of ANP-3904P presents the disposition status of each of the Browns Ferry FSAR Chapter 14 transient events, with respect to thermal limit response, and is not related to potentially limiting ASME overpressurization events which are described in FSAR Section 4.4.6. The primary difference between the MSIV closure described in FSAR Section 14.5.2.7 and the event defined in FSAR Section 4.4.6 is that the event defined in Section 14.5.2.7 credits the scram signal on the MSIV position, whereas the overpressurization event defined in Section 4.4.6 explicitly assumes that this scram signal fails. Allowing credit for the scram signal on MSIV position greatly reduces the severity of the event, which is the basis for the determination that the MSIV closure event of Section 14.5.2.7, with respect to thermal limit response, is bounded by the LRNB with EOC-RPT-OOS of Section 14.5.2.2.4.

The ASME event with MSIV closure presented in Section 4.1.4 of ANP-3904P provides a demonstration of the AURORA-B AOO methodology to the ASME overpressurization

event consistent with the description of events provided in FSAR Section 4.4.6. Historically, with Framatome methodology applied at Browns Ferry, the ASME-MSIV closure event results in the highest peak pressure compared to an ASME-TCV or ASME-TSV closure event. However, these three valve closures are considered potentially limiting and are analyzed on a cycle-specific basis to confirm the pressure limits are supported for operation. Section 3.2 of ANP-3904P discusses the Framatome approach for developing the cycle-specific calculation plan, which will identify the necessary analyses to ensure that all potentially limiting events will be appropriately evaluated.

SNSB RAI 8:

Regulatory Basis:

The Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP, NUREG-0800), Section 4.2, “Fuel System design” (ADAMS Accession No. ML070740002), Section 4.3, “Nuclear Design”, and Section 4.4, “Thermal and Hydraulic Design” (ADAMS Accession No. ML070740003), provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and thermal and hydraulic design of the core.

According to SRP Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and Anticipated Operational Occurrences (AOOs)
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and Coolability is always maintained.

1967 Atomic Energy Commission (AEC) CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A) – See SNSB RAI 1.

Request:

ANP-3860P defines criteria for fuel assembly lift-off as, “The fuel shall not levitate under normal operating or AOO conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired.”

(a) Provide a summary of key steps in calculations of assembly lift-off during normal operating conditions for both ATRIUM 11 core and mixed core conditions.

Response:

In summary, the key steps to ensure the fuel does not separate from the fuel support during normal operation for a full core of ATRIUM 11 fuel assemblies or a mixed core are listed below.

1) [[ ]]
2) The downward forces are summed. The downward forces include the fuel assembly weight, the weight of the fluid inside the fuel channel [[ ]] and the downward effect due to a change in momentum of the fluid inlet and outlet flow rate.
3) The upward forces are calculated based on the fuel assembly inlet and bypass pressure differential provided by [[ ]]
4) [[ ]]
5) [[ ]] and mixed core of the ATRIUM 11 and co-resident fuel or full core of the ATRIUM 11 fuel assembly.
6) Liftoff conditions are confirmed each Framatome fuel assembly reload.

(b) For faulted or accident conditions, such as a LOCA, provide a summary of procedures with a typical calculation describing how the criteria for assembly lift-off is satisfied.

Response:

For faulted or accident conditions, the ATRIUM 11 fuel assembly [[ ]]

[[

]] The fuel is confirmed to not disengage from the fuel assembly support piece during faulted or accident conditions and is verified each reload of Framatome fuel.

SNSB RAI 9:

Regulatory Basis:

See SNSB RAI 8.

Request:

With regard to rod bow, Section 3.3.5 of ANP-3860P states that **[[**

]]

Provide details of how this correlation is developed. Also describe how this correlation is used to quantify the creep as a function of fuel exposure.

Response:

The full description is given in BAW-10247P-A, Supplement 2P-A, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," Framatome Inc., Reference 8 Section 4.1.5.1 and Appendix A. Framatome uses an empirical model to quantify the creep versus exposure.

SNSB RAI 10:

Regulatory Basis:

See SNSB RAI 8.

Request:

Section 3.1 of ANP-3866P states, **[[**

]]

(a) Describe the neutronic impact, if any, of Cr in the fuel.

Response:

This item has been addressed in the approved topical report ANP-10340P-A (Reference 5) as described below.

[[

]]

(b) Describe the impact of Cr in fuel on fission gas release, fuel densification and swelling, corrosion, and fuel creep.

Response:

The fuel thermal-mechanical processes mentioned in the question, together with all other material properties have been addressed in ANP-10340P-A (Reference 5), as follows:

- **[[**

]]

- [[

]]

SNSB RAI 11:

Regulatory Basis:

See SNSB RAI 8.

Request:

Section 3.2 of ANP-3866P describes application of RODEX4 and statistical methodology for thermal-mechanical response of the fuel rod surrounded by coolant. Provide the following information:

(a) Explain how **[[**

]]

Response:

The radial depression of the thermal flux is one component of the radial power profile model of RODEX4. **[[**

]] The volumetric thermal power at any location in the fuel rod is the product of the value of the radial power profile factor at that radius, the input linear power at the axial location and the volume of **[[**

]]

(b) Explain the term **[[** **]]**

Response:

Neutronic fuel assembly typing is an identification scheme that groups fuel assemblies by the enrichment and gadolinia distribution within the fuel rods that comprise the assembly. Thus, all fuel assemblies within a given type have the same distribution of these two characteristics. For a given type, the mechanical fuel assembly designs are identical (e.g., number of fuel rods; number, location, and length of part-length fuel rods; plenum volumes for each fuel rod; spacer grid design, water channel design, etc.). A reload batch of BWR fuel usually consists of fuel assemblies with identical mechanical designs, but with two or three neutronic types—and occasionally more.

(c) Explain how **[[** **]]** are calculated.

Response:

[[

]]

(d) Describe the methodology used for power measurement and operational uncertainties, manufacturing uncertainties, and model uncertainties. Provide a summary of these uncertainties.

Response:

[[

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SNSB RAI 12:

Regulatory Basis:

See SNSB RAI 8.

Request:

Section 3.3.7 of ANP-3866P states that **[[**
]] at Browns Ferry.
Provide details of how this limit is implemented at Browns Ferry.

Response:

[[

]]

SNSB RAI 13:

Regulatory Basis:

Same as in SNSB RAI 1

Request:

In the NRC staff SE for ANP-10332P-A, limitation and condition #16 states that plant licensing applications referencing the AURORA-B LOCA Evaluation Model shall justify that the input conditions assumed in the analysis are bounding across the entire approved operating domain.

[[

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Response:

Section 4.3.1.3 of Reference 9 provides a discussion for the necessary statepoints for supporting LOCA analysis within the MELLLA+ boundary. Framatome LOCA calculations were performed for the **[[**

]] in order to support the acceptance criteria of 10 CFR 50.46 within the MELLLA+ boundary. The approach used for off-rated statepoint evaluations in the ATRIUM 11 LOCA analysis is consistent with that previously approved for the Browns Ferry MELLLA+ LAR, Reference 10, Section 3.4.3. Results for these statepoints are summarized in ANP-3905P for ATRIUM 11 fuel and demonstrate compliance with Limitation and Condition #16 of ANP-10332P-A.

3.0 REFERENCES

1. NRC Electronic Mail to TVA, “Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Request for Additional Information re LAR to Use Advanced Framatome Methodologies in Support of ATRIUM 11 Fuel (EPID L-2021-LLA-0132),” dated June 3, 2022 (ML22160A474 and ML22160A681).

2. ANP-10335P-A, Revision 0, “ACE/ATRIUM 11 Critical Power Correlation,” Framatome Inc., May 2018.

3. ANP-10307PA, Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” AREVA NP Inc., June 2011.

4. ANP-10300P-A, Revision 1, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,” Framatome Inc., January 2018.

5. ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods,” Framatome Inc., May 2018.

6. BAW-10247PA, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” AREVA NP Inc., February 2008.

7. ANP-3866P, Revision 0, “ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry LAR,” Framatome Inc., October 2020.

8. BAW-10247P-A Supplement 2P-A Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods,” Framatome Inc., August 2018.

9. Safety Evaluation by the Office of Nuclear Reactor Regulation, Licensing Topical Report NEDC-33006P, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” General Electric Hitachi Nuclear Energy America, LLC, October 2008. (ML08113008)

10. Letter, F. Saba (USNRC) to J. Barstow (TVA), “Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendment Nos. 310, 333, and 293 Regarding Maximum Extended Load Line Limit Analysis Plus (EPID L-2018-LLA-0048). ADAMS Accession Number ML 19210C308.

11. XN-NF-82-07(P)(A) Revision 1, “Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model,” Exxon Nuclear Company, November 1982.

Enclosure 3

Framatome Affidavit for Enclosure 1

A F F I D A V I T

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-4006P, Revision 1 "Browns Ferry Advanced Methods License Amendment Request – Response to Request for Additional Information," dated July 2022 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 13, 2022

MEGINNIS Alan Digitally signed by MEGINNIS Alan
Date: 2022.07.13 08:45:56 -07'00'

Alan B. Meginnis