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

December 6, 2018

Mr. Joseph W. Shea
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT PLUS LICENSE AMENDMENT REQUEST (EPID L-2018-LLA-0048)

Dear Mr. Shea:

By letter dated February 23, 2018, as supplemented by letters dated March 7, and July 23, 2018, Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. The proposed LAR would allow operation of BFN units in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain and use of the Detect and Suppress Solution - Confirmation Density stability solution.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's submittal and determined that additional information is needed. A draft request for additional information (RAI), questions, and inputs from NRC staff in the Office of Nuclear Reactor Regulation, Division of Safety Systems, Reactor Systems Branch (SRXB) and Nuclear Performance and Code Branch (SNPB) were forwarded by electronic mail (e-mail) to TVA in September 2018. The NRC staff discussed with the TVA staff the draft RAIs during a regulatory audit from October 9 to 11, 2018, at the Excel Services Corporation in Rockville, MD. The NRC staff's finalized RAIs are provided in Enclosures 1, 2, 3, and 4.

The NRC staff has determined that the documented RAIs in Enclosures 1 and 3 contain proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has prepared redacted, non-proprietary versions in Enclosure 2 and 4.

Enclosures 1 and 3 transmitted herewith contain Sensitive Unclassified Non-Safeguard Information. When separated from Enclosures 1 and 3, this document is decontrolled.

By an e-mail dated November 21, 2018, Mr. Daniel Green of your staff proposed to submit TVA's RAI responses by Jaunary 18, 2019. The NRC staff agreed with TVA's proposed response date.

If you have any questions, please contact me at 301-415-1447 or Farideh.Saba@nrc.gov.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260,
and 50-296

Enclosures:
1. RAIs from SRXB (Proprietary)
2. RAIs from SRXB (Non-Proprietary)
3. RAIs from SRXB – Containment (Proprietary)
4. RAIs from SRXB – Containment (Non-Proprietary)

cc w/o Enclosures 1 and 3: Listserv

ENCLOSURE 2

NON-PROPRIETARY

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF SAFETY SYSTEMS

TO TENNESSEE VALLEY AUTHORITY

REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT PLUS

LICENSE AMENDMENT REQUEST FOR

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF SAFETY SYSTEMS

TO TENNESSEE VALLEY AUTHORITY

REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT PLUS

LICENSE AMENDMENT REQUEST FOR

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

BACKGROUND

By letter dated February 23, 2018 (Reference 1), as supplemented by letters dated March 7 (Reference 2), and July 23, 2018 (Reference 3), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. The proposed LAR would allow operation of BFN units in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain and use of the Detect and Suppress Solution - Confirmation Density (DSS-CD) stability solution.

REGULATORY BASIS

The regulatory bases for the U.S. Nuclear Regulatory Commission (NRC) staff review are the requirements contained in Title 10 of the *Code of Federal Regulations* (10 CFR) as follows:

- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," insofar as they establish the requirements and acceptance criteria for emergency core cooling system (ECCS) design, and for the evaluation models (EMs) used to evaluate ECCS performance during a hypothetical loss-of-coolant (LOCA). Specific considerations include:
    - 10 CFR 50.46(a)(1)(i) requires the use of an acceptable EM to evaluate ECCS performance under the conditions of a hypothetical LOCA, and 10 CFR 50.46(a)(1)(ii) allows for the development of an EM that conforms to the required and acceptable features specified in Appendix K to 10 CFR Part 50.
    - 10 CFR 50.46(a)(1)(i) also requires ECCS cooling performance to be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe hypothetical LOCAs are calculated.
    - Acceptance criteria set forth in paragraph (b) of 10 CFR 50.46, and the results of the ECCS evaluation must show that the acceptance criteria are met. Among

others, these include requirements related to peak cladding temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation.

- 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” which requires, in part that:

  (1) Each boiling-water reactor (BWR) have an alternate rod injection system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

  (2) Each BWR have a standby liquid control system with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gallons per minute (gpm) of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel.

  (3) Each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. ATWS is defined as an anticipated operation occurrence (AOO) followed by the failure of the reactor trip portion of the protection system.

In addition to the 10 CFR 50.62 requirements, the NRC staff reviewed the licensee's ATWS analysis to ensure that the following ATWS acceptance criteria are met. These acceptance criteria are adopted by the licensee in Section 9.3.1 of the MELLLA+ Safety Analysis Report (M+SAR), NEDC-33877P, Revision 0, for proprietary and NEDO-3377NP for non-proprietary versions (Reference 4):

- Maintain reactor vessel integrity

- Maintain containment integrity

- Maintain coolable core geometry

The BFN units were designed and constructed based on the proposed general design criteria (GDC) published by the Atomic Energy Commission in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereafter called “draft GDC”). The following draft GDC were used as a regulatory basis, as applicable, for the requests for additional information (RAIs):

- Draft GDC 6: 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.

- Draft GDC 7: The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

- Draft GDC 32: Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Each of the RAIs below draws on the basis identified above, and as appropriate, some RAIs further reference additional regulatory requirements and guidance.

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

The Nuclear Regulatory Commission NRC staff from the Office of Nuclear Reactor Regulation, Division of Safety System, Reactor System Branch (SRXB) and Nuclear Performance and Code Review Branch (SNPB) reviewed the licensee's submittals and determined that the following RAI is needed to complete its review.

The NRC staff determined that the following RAI contains proprietary information pursuant to 10 CFR 2.390. Proprietary information is identified by bold text enclosed within double brackets, as shown here **[[example proprietary text]]**.

<u>SRXB RAI-1</u>

Section 9.3.3 of the M+SAR, NEDC-33877P, Revision 0 contains an ATWS with core instability (ATWS-I) sensitivity study that contains six fuel related parameters which are varied to determine their impact on the analysis results. However, the licensing basis ATWS analysis in Section 9.3.1.1 of the M+SAR contains three of the same parameters for sensitivity studies but does not include **[[ ]]**. Explain why these sensitivity studies are not necessary to be completed for the ATWS analysis to ensure that they are not necessary to demonstrate compliance with the ATWS acceptance criteria.

<u>SRXB RAI-2</u>

The ATWS-I fuel parameter sensitivity studies in Section 9.3.3 of the M+SAR were completed for the 2-recirculation pump trip (2RPT) event. Provide the results of the ATWS-I fuel parameter sensitivity studies for the **[[ ]]** case (equivalent to Table 9-11 in the M+SAR) to ensure that the 2RPT event will continue to be limiting and to demonstrate that the ATWS acceptance criteria are met for potentially limiting ATWS-I events.

<u>SRXB RAI-3</u>

The ATWS-I sensitivity study using homogeneous nucleation for minimum stable film boiling temperature (T_{min}) was provided for the 2RPT event in Section 9.3.3 of the M+SAR. Please provide the result using homogeneous nucleation T_{min} model for the turbine trip with bypass

(TTWBP) case to ensure that the 2RPT event will continue to be limiting and to demonstrate that the ATWS acceptance criteria are met for potentially limiting ATWS-I events.

SRXB RAI-4

Table 9-10, "PCT Results for ATWSI Sensitivity Analysis," of the M+SAR (Reference 4) shows that the bounding fuel parameter sensitivity case for **[[** **]]** gives an **[[** **]]** in PCT compared to using nominal fuel parameter values; however, for the Homogeneous Nucleation case, the bounding fuel parameter sensitivities account for **[[** **]]** increase in PCT over the nominal fuel parameter case. To ensure the model adequately models the transient to meet the ATWS acceptance criteria, explain why this larger PCT increase occurred for the Homogeneous Nucleation case, using detailed TRACG results from relevant ATWS-I cases to support the explanation.

SRXB RAI-5

Attachment 33 of the LAR contains the basis for feedwater temperature reduction for input into the TRACG ATWS-I analysis. The data was based on a turbine trip event at BFN, Unit 3. To ensure the model adequately models the transient to meet the ATWS acceptance criteria:

f. Provide the basis for the 39,800 pounds mass (lbm) used in the calculation which determined the 14 seconds delay time.

g. Justify that the 14 seconds delay time is bounding relative to the turbine trip event data for BFN, Unit 3.

h. Explain why the feedwater temperature reduction rate changes during the turbine trip event at BFN, Unit 3 (Step 1, Step 2, and Step 3), and please discuss why this is bounding for ATWS-I.

i. Please compare the TRACG feedwater temperature input to ATWS simulator data to demonstrate that the feedwater temperature used in the analysis is conservative.

j. Please provide justification that this basis is also applicable to the feedwater temperature reduction for the 2RPT event

SRXB RAI-6

The M+SAR justifies the use of DSS-CD with ATRIUM 10XM fuel. Justify that ATRIUM 10 fuel, which will also be present in the core when MELLLA+ is implemented, is bounded by ATRIUM 10XM fuel such that no explicit analysis for ATRIUM 10 is necessary to ensure both fuels meet draft GDC 6 and 7.

SRXB RAI-7

Section 2.4.1 of the M+SAR states that a **[[** **]]** for DSS-CD will be used for BFN. To ensure the analysis meets draft GDC 6 and 7, provide the following:

c. The minimum period for all the cases analyzed.

d. Justification that the cases analyzed produced the minimum expected period in the ranges where DSS-CD will be applied.

SRXB RAI-8

Provide the key parameter figures (like those in ANP-3552 (Reference 5) Figures 5.1 through 5.6) for the 100 percent power and 85 percent core flow case for load reject no-bypass and the feedwater controller failure to understand the impact of MELLLA+ on these analyses to ensure they meet draft GDC 7. Discuss if the AOOs are reanalyzed each reload for all the MELLLA+ domain statepoints, and if not, describe why reanalysis is not necessary and what parameters are checked to ensure the original cases remain bounding (i.e., describe why the bounding analyses first analyzed for MELLLA+ will remain bounding in future cycles).

SRXB RAI-9

10 CFR 50.46(a)(1)(i) requires, in part, that the ECCS cooling performance must be calculated in accordance with an acceptable EM and must be calculated for postulated LOCA with different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. To ensure that the most severe postulated LOCAs are calculated, provide a sensitivity study for a point between 85 percent flow and 99 percent flow at full power for the limiting break size provided in the ANP-3546 (Reference 6), "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU MELLLA+)."

SRXB RAI-10

Section 2.0 of ANP-3546 discusses that the break spectrum analysis was completed for ATRIUM 10XM fuel and justifies that it is applicable to other co-resident fuel, since their thermal-hydraulic characteristics are similar. Confirm that the only co-resident fuel during BFN MELLLA+ operation is ATRIUM 10. If not, provide additional justification for the other fuel types to ensure all fuel types are covered to meet the 10 CFR 50.46 acceptance criteria.

SNPB RAI-1

Section 9.3.1.1 of the M+SAR (Reference 4) states that **[[**

]]. To ensure the model adequately models the transient to meet the ATWS acceptance criteria, provide the following:

e. Explain how the GEXL97 correlation is used in the ATWS-I analysis; especially during the dryout and rewet stages during an ATWS-I event (during oscillatory behavior).

f. Explain how the GEXL97 coefficients are determined for ATRIUM 10XM (used in ATWS-I and DSS-CD calculations).

g. Provide a summary of how the R-factors associated with GEXL97 correlation are determined for ATRIUM 10XM (used in the ATWS-I and DSS-CD calculations).

h. Provide a summary of how the fuel rod location and geometry dependent additive constants are determined for the R-factors.

SNPB RAI-2

ANP-3544 (Reference 7) and ANP-3568 (Reference 8) state that the equilibrium cycle assumes the use of blended low enriched uranium material for one fuel type to account for about 30 percent of the fresh reload assemblies. Discuss the impact of the use of blended low enriched uranium fuel during the MELLLA+ operation to ensure that it is appropriately accounted for in the steady state, transient, and accident analyses.

SNPB RAI-3

ANP-3544 states that the core hot excess reactivity was calculated at full power with all rods out, 102.5 mega pounds per hour (Mlb/hr) core flow, with equilibrium xenon. Discuss whether the hot excess reactivity calculated at this condition (full power, all rods out, 102 Mlb/hr) is suitable and valid for the BFN MELLLA+ operating domain to ensure the shutdown margin was appropriately determined for MELLLA+ conditions.

SNPB RAI-4

ANP-3550 (Reference 9) Section 3.0 states that for the control rod drop analysis (CRDA) the deposited enthalpy must be < 280 calories per gram (cal/g), which is used to demonstrate compliance with draft GDC 32. Since the publication of Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Reference 10), the acceptance criteria of 280 cal/g has been determined to be inadequate to ensure fuel rod geometry and long-term coolability. The NRC staff documented its position on RG 1.77 in a letter dated April 3, 2015, "Results of Periodic Review of Regulatory Guide 1.77" (Reference 11). The position is supported by a guidance document dated Jaunary 19, 2007, titled "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance" (Reference 12).

An analysis for demonstrating acceptance to draft regulatory guide (DG) 1327 is provided in ANP-3633 (Reference 13), however, the LAR states that the DG-1327 is not included in the BFN licensing basis. Since the staff has determined that the 280 cal/g is non-conservative and since it is stated that DG-1327 is not part of the licensing basis, discuss what acceptance criteria will be used if it is necessary to reanalyze the CRDA event (e.g., an error is found in the analysis).

<u>REFERENCES</u>

1 TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus," dated February 23, 2018 (ADAMS Accession No. ML18079B140).

2 TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus, Supplement 1," dated March 7, 2018 (ADAMS Accession No. ML18067A493).

3 TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus,

Supplement 2, Operator Training Results," dated July 23, 2018 (ADAMS Accession No. ML18205A498).

4 NEDO-33877, Revision 0 (Attachment 6 of LAR), "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Maximum Extended Load Line Limit Plus," dated February 2018 (ADAMS Accession No. ML18079B140).

5 ANP-3552NP, Revision 0 (Attachment 18 of LAR), "Browns Ferry Unit 3 Cycle 19 Representative Reload Analysis (EPU MELLLA+)," Dated December 2017 (ADAMS Accession No. ML18079B140).

6 ANP-3546NP, Revision 0 (Attachment 12 of LAR), "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU MELLLA+)," dated March 2017 (ADAMS Accession No. ML18079B140).

7 ANP-3544NP, Revision 0 (Attachment 10 of LAR), "Browns Ferry EPU (120% OLTP) MELLLA+ Equilibrium Fuel Cycle Design," dated December 2017 (ADAMS Accession No. ML18079B140).

8 ANP-3568NP, Revision 2 (Attachment 22 of LAR), "Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate/MELLLA+," dated February 2018 (ADAMS Accession No. ML18079B140).

9 ANP-3550NP, Revision 0 (Attachment 26 of LAR), "Evaluation of AREVA Fuel Thermal-Hydraulic Performance for Browns Ferry at EPU MELLLA+," dated March 2017 (ADAMS Accession No. ML18079B140).

10 U. S. RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," dated May 1974 (ADAMS Accession No. ML003740279).

11 U.S. NRC Memorandum, "Results of Periodic Review of Regulatory Guide 1.77," dated April 3, 2015 (ADAMS Accession No. ML15075A311).

12 U.S. NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance," dated January 19, 2007 (ADAMS Accession No. ML070220400).

13 ANP-3633NP, Revision 1 (Attachment 32 of LAR), "Browns Ferry EPU MELLLA+ CRDA Assessment with DG 1327 Criteria," dated January 2018 (ADAMS Accession No. ML18079B140).

ENCLOSURE 4

NON-PROPRIETARY

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF SAFETY SYSTEMS

TO TENNESSEE VALLEY AUTHORITY

REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT PLUS

LICENSE AMENDMENT REQUEST FOR

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

REQUEST FOR ADDITIONAL INFORMATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF SAFETY SYSTEMS
TO TENNESSEE VALLEY AUTHORITY
REGARDING CONTAINMENT SECTION OF
MAXIMUM EXTENDED LOAD LINE LIMIT PLUS
LICENSE AMENDMENT REQUEST FOR
BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3
DOCKET NOS. 50-259, 50-260, AND 50-296

BACKGROUND

By letter dated February 23, 2018 (Reference 1), as supplemented by letters dated March 7 (Reference 2), and July 23, 2018 (Reference 3), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for an amendment to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. The proposed LAR would allow operation in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain and use of the Detect and Suppress Solution - Confirmation Density stability solution.

The U.S. Nuclear Regulatory Commission (NRC) staff from the Office of Nuclear Reactor Regulation, Division of Risk Assessment, from PRA [Probabilistic Risk Assessment] Operations and Human Factors Branch (APHB) reviewed the containment related portions in Section 4.0, "Engineered Safety Features" in Enclosure 5 or MELLLA+ Safety Evaluation Report (M+SAR) (Reference 2) and determined that the following request for additional information (RAI) is needed to complete its review.

The NRC staff from the Office of Nuclear Reactor Regulation, Division of Safety System, Reactor System Branch (SRXB) reviewed the licensee's submittals associated with containment analyses (SRXB-C) and determined that the following RAI is needed to complete its review.

The NRC staff determined that the following RAI contains proprietary information pursuant to Title 10 of *Code of Federal Regulations* Section 2.390. Proprietary information is identified by bold text enclosed within double brackets, as shown here **[[example proprietary text]]**.

The BFN units were designed and constructed based on the proposed general design criteria (GDC) published by the Atomic Energy Commission in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereafter called "draft GDC"). The draft GDC were used as a regulatory basis, as applicable, for the following RAIs:

SRXB-C RAI-1

10 CFR 50.49(e)(1) states that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident (DBA) during or following which this equipment is required to remain functional.

Section 4.1.1 of MELLLA+ Safety Analysis Report (M+SAR), NEDC-33877P, Revision 0, for proprietary and NEDO-3377NP for non-proprietary versions (Reference 4) states that the current long-term analysis for small steam line break (SSLB) accident for evaluation of drywell equipment qualification produced a significantly high peak drywell temperature of 336.9 degrees Fahrenheit (°F) and an elevated drywell atmosphere temperature response that lasts for a much longer duration than produced by the short-term recirculation suction line break accident analysis. It is further stated that the peak predicted drywell shell temperature produced by the current SSLB analyses of 280.8°F is bounded by the drywell shell design limit of 281°F. Provide the results of the drywell gas temperature response and the peak drywell shell temperature for the SSLB accident in the MELLLA+ operating domain.

SRXB-C RAI-2

Draft GDC 10 and 49, as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA).

Section 4.1.1.1 of M+SAR (Reference 4) states:

> In addition, there is no change as a result of the MELLLA+ operating domain expansion to other key long-term containment response parameters reported in Reference 16 [Reference 3 of this document] including drywell atmosphere temperature and drywell shell temperature response, wetwell temperature and wetwell pressure response, and steam bypass capability. No further evaluation of these long-term containment response parameters is therefore required for MELLLA+.

Provide reasons why the parameters stated above are not affected by the MELLLA+ operating domain. In case the reason is the long-term decay heat is not changed in the MELLLA+ operating domain, justify these parameters solely depend on decay heat.

SRXB-C RAI-3

Draft GDC 40 and 42, insofar as they require that protection be provided for engineered safety features against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA.

In order to meet the above requirement of draft GDC 40 and 42, it is necessary to assure that the vent thrust loads, which is one of the categories of dynamic loads imposed on the containment and its internal SSCs, during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.2.1 of M+SAR, under the heading "Vent Thrust Loads" states:

> Vent thrust loads are calculated using the equations documented in the LDR [load definition report) (Reference 31 [Reference 6 of this document]) at MELLLA+ conditions, based on the DBA-LOCA results obtained with the GEH [General Electric Hitachi] M3CPT code.

Explain how the MELLLA+ vent thrust loads were calculated based on the DBA-LOCA results using the equations in Reference 6, and which DBA-LOCA results were used.

SRXB-C RAI-4

Draft GDC 40 and 42, insofar as they require that protection be provided for engineered safety features against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA.

In order to meet the above requirement of draft GDC 40 and 42, it is necessary to assure that the pool swell loads, which is one of the categories of dynamic loads imposed on the containment and its internal SSCs, during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.2.1 of M+SAR, under the heading "Pool Swell Loads" states:

> [[
>
>]]

As stated in Section 2.6.1 of NEDC-33860P, Revision 1, "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate" (non-proprietary version in Reference 5), the current short-term drywell pressure response (drywell pressure versus time) was calculated using the M3CPT code which also provides the drywell pressurization rate. Explain what other method was used for calculating the drywell pressurization rate at the MELLLA+ condition and how the results compare with the M3CPT results.

SRXB-C RAI-5

Draft GDC 10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain functional capability for as long as the situation requires.

Draft GDC 41 and 52 in part requires a system to remove heat from the reactor containment shall be provided.

Section 4.2.6.1 of M+SAR, last sentence in sixth paragraph states:

> The RHR [residual heat removal] pump flow used in the ATWS [anticipated transient without scram] NPSH [net positive suction head] analysis was

increased by a factor of 1/√0.97 (1.015) to account for the reduction in pump flow rate associated with a 3% reduction in pump total developed head.

Please clarify.

SRXB-C RAI-6

Regulatory basis in SRXB-C-RAI 5 is applicable to this RAI.

Table 4-6 of M+SAR (Reference 4), Note 2 is not clear. It states that the MELLLA+ "ATWS non-LOOP [loss of offsite power] PRFO EOC [End of Cycle]" results for peak suppression pool temperature are lower than the extended power uprate (EPU) results due to the use of more current ATWS analysis. The NRC staff requests details on the more current ATWS modeling and its differences/comparison with the EPU modeling, including the methodologies used. For reference, following are the licensing basis parameters for the EPU ATWS containment and NPSH analysis in Table 2.6.5-3 in the NRC Safety Evaluation dated August 14, 2017 (Reference 7).

Special Event	RHR Flow per HX* (gpm) (Note 1)	Initial SP** Temp (°F)	Peak SP Temp (°F)	RHRSW		Number of HXs used	K-value for 1 RHR HX (BTU/second-°F)
				Flow per RHR HX (gpm)	Temp (°F)		
ATWS-LOOP	6500	95	173.3	4500	95	2	277
ATWS-non LOOP	6500	95	171.8	3800	95	4	259
Note 1: Safety analysis flows assumed used for determining suppression pool temperature * HX – Heat Exchanger ** SP – Suppression Pool							

REFERENCES

8. TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus," dated February 23, 2018 (ADAMS Accession No. ML18079B140).

9. TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus, Supplement 1," dated March 7, 2018 (ADAMS Accession No. ML18067A493).

10. TVA letter to the U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus, Supplement 2, Operator Training Results," dated July 23, 2018 (ADAMS Accession No. ML18205A498).

11. NEDO-33877, Revision 0 (Attachment 6 of LAR), "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Maximum Extended Load Line Limit Plus," dated February 2018 (ADAMS Accession No. ML18057B276).

12. GE Hitachi Nuclear Energy, NEDO-33860NP, Revision 1 "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate," dated October 2016 (ADAMS Accession No. ML16302A441).

13. GE Nuclear Energy, NEDO-21888, Revision 2, "Mark I Containment Program Load Definition Report," November 1981.

14. U.S. NRC letter to TVA, Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Extended Power Uprate, dated August 14, 2017 (ADAMS Accession No. ML17032A120).

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT PLUS LICENSE AMENDMENT REQUEST (EPID L-2018-LLA-0048) DATED DECEMBER 6, 2018

DISTRIBUTION:
NON-Public
PM File Copy
RidsNrrDorlLpl2-2
RidsNrrLABClayton
RidsRgn2MailCenter
RidsNrrPMBrownsFerry
RidsNrrDssSrxb
RidsNrrDssSnpb
MPanicker, NRR
ASallman, NRR
RidsACRS_MailCTR
JBorromeo, NRR
DWoodyatt, NRR

ADAMS Accession Nos.:
ML18331A546 (Proprietary)
ML18331A544 (Non-Proprietary)

* via email

OFFICE	DORL/LPLII-2/PM	DORL/LPLII-2/LA	DSS/SRXB/BC*
NAME	FSaba	BClayton (IBetts for)	JWhitman
DATE	12/06/18	12/06/18	11/29/18
OFFICE	DSS/SRXB/BC* (Containment)	DORL/LPLII-2/BC	DORL/LPLII-2/PM
NAME	JWhitman	UShoop	FSaba
DATE	11/15/18	12/06/18	12/06/18

OFFICIAL RECORD COPY