Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-18-042

March 7, 2018

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: **Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus, Supplement 1**

Reference: TVA Letter to NRC, CNL-18-002, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus," dated February 23, 2018 (ML18057B276)

By the reference letter, Tennessee Valley Authority (TVA) submitted a request for a Technical Specification (TS) amendment (TS-510) 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 license amendment request (LAR) allows operation 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.

The purpose of this letter is to provide BFN-related responses to specific Brunswick Steam Electric Plant (BSEP) requests for additional information (RAIs) that were sent by NRC to Duke Energy. Enclosure 1 provides General Electric - Hitachi Nuclear Energy Americas LLC (GEH) information responsive to specific RAIs applicable to the BFN MELLLA+ LAR. GEH consider portions of the information provided in Enclosure 1 of this supplement to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390. An affidavit for withholding information, executed by GEH, is provided in Enclosure 5. A non-proprietary version of the document is provided in Enclosure 2. Therefore, on behalf of GEH, TVA requests that Enclosure 1 be withheld from public disclosure in accordance with the GEH affidavit and the provisions of 10 CFR 2.390.

Enclosure 3 provides Framatome, Inc. (Framatome) information responsive to specific BSEP RAIs applicable to the BFN MELLLA+ LAR. Framatome considers portions of the information provided in Enclosure 3 of this supplement to be proprietary and, therefore exempt from public disclosure pursuant to 10 CFR 2.390. An affidavit for withholding information, executed by Framatome, is provided in Enclosure 6. A non-proprietary version of the document is provided in Enclosure 4. Therefore, on behalf of Framatome, TVA requests that Enclosure 3 be withheld from public disclosure in accordance with the Framatome affidavit and the provisions of 10 CFR 2.390.

TVA has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in the reference letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the supplemental information in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed license amendment. 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 State Department of Public Health.

There are no new regulatory commitments associated with this submittal. If there are any questions or if additional information is needed, please contact Edward Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 7th day of March 2018.

Respectfully,

J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosures:

1. Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+ ATWS Supplemental Information (proprietary)
2. Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+ ATWS Supplemental Information (non-proprietary)
3. Framatome Additional Information for Browns Ferry MELLLA+ (proprietary)
4. Framatome Additional Information for Browns Ferry MELLLA+ (non-proprietary)
5. Affidavit – GE-Hitachi Nuclear Energy Americas LLC
6. Affidavit – Framatome Inc.

cc: See Page 3

ENCLOSURE 1

Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+
ATWS Supplemental Information
(Proprietary)

ENCLOSURE 2

Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+

ATWS Supplemental Information

(Non-proprietary)

Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+ ATWS Supplemental Information

Note: This is a non-proprietary version of the document 004N6892-P Revision 1 which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

1.0 Introduction

In support of the Brunswick Steam Electric Plant (BSEP) License Amendment Request (LAR) for an expansion of the core power flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)), the Nuclear Regulatory Commission (NRC) requested additional information to facilitate the review. Reference 1 presents the requested information.

This document presents information responsive to the applicable requests for additional information for the Browns Ferry Nuclear Plant (BFN) which supports the BFN MELLLA+ LAR.

2.0 TRACG Modeling

2.1. LHGR

Steady-state conditions consistent with the equilibrium cycle conditions provided by Framatome (formerly AREVA) were used to initiate the TRACG Anticipated Transient Without Scram (ATWS) analyses. A hot rod is set for each TRACG channel group. A hot rod is modeled with an additional rod group in the channel component selected to represent the hot bundle (highest radial power, highest harmonic, and highest hybrid power at symmetrical section of the core). The hot rod (i,j) location in a bundle is [[ ]]. The hot rod relative power, CPOWR in TRACG, is selected such that in steady-state at the intended core state point the peak Linear Heat Generation Rate (LHGR) of the hot rod is no less than [[ ]] of the maximum LHGR limit. The CPOWR inputs determined for the hot rod are applied to all other TRACG channel groups. [BSEP MELLLA+ SRXB-RAI-4a (Reference 1)]

2.2. Maximum Time Step

A maximum time step (DTMAX) of [[ ]] was used in the TRACG Recirculation Pump Trip (RPT) and Turbine Trip With Bypass (TTWBP) analyses presented in the MELLLA+ Safety Analysis Report (M+SAR) and is consistent with the GE Hitachi Nuclear Energy (GEH) ATWS Instability (ATWSI) process. This value is [[ ]]. [BSEP MELLLA+ SRXB-RAI-4b (Reference 1)]

2.3. Channel Grouping

Figures 1 and 2 show the diagrams of the TRACG channel grouping used for the regional and core-wide modes for RPT and TTWBP ATWSI analyses at Middle-of-Cycle (MOC). The regional mode analyses result in a higher Peak Cladding Temperature (PCT) value than the core-wide mode analyses. [BSEP MELLLA+ SRXB-RAI-5a (Reference 1)]

2.4. R-Factor

The R-factor used for all TRACG channels is [[ ]] and is within the GEXL97 correlation range for ATRIUM-10. PCT results from the sensitivity analyses performed with the GEXL critical power uncertainty range [[ ]] show that PCT effect is insignificant (See M+SAR Table 9-11). Therefore, no additional R-factor sensitivity studies are necessary because the sensitivities are already covered by the [[ ]] critical power sensitivity studies. [BSEP MELLLA+ SRXB-RAI-10 (Reference 1)]

2.5. Application of GEXL to Oscillatory Conditions

The GEXL correlation is used to determine the occurrence of boiling transition. For TRACG ATWSI calculations, this is no different from its planned use and is consistent with the approved Detect and Suppress Solution – Confirmation Density (DSS-CD) application with TRACG during oscillatory conditions (Reference 2). GEXL correlations developed for Global Nuclear Fuel (GNF) fuel products since GE9 have been assessed against oscillatory test data to confirm their performance in predicting the onset of boiling transition and the return to nucleate boiling. In this analysis, the GEXL correlation for ATRIUM-10XM fuel is provided by TVA / Framatome (AREVA) in a form that is compatible with TRACG. [BSEP MELLLA+ SRXB-RAI-9a and BSEP MELLLA+ SNPB-RAI-4c (Reference 1)]

2.6. Critical Heat Flux Determination in Oscillatory Phase

Section 6.6.6 of the TRACG Model Description (Reference 3) describes the correlations used to determine when transition boiling occurs and how a rod returns to nucleate boiling. When GEXL applies (See Section 6.6.6 of Reference 3), transition boiling occurs [[

]]. Film boiling will always occur in TRACG if the minimum stable film boiling temperature (T_{min}) is exceeded regardless of the critical heat flux temperatures and critical qualities. Section 6.6.8 of the TRACG Model Description (Reference 3) describes the interpolation of heat transfer coefficients (HTCs) when the clad surface temperature is above T_{CHF} and below T_{min}.

The GEXL correlation does not determine the HTCs. The GEXL correlation is utilized for determining the boundaries between correlations that are used to calculate HTCs. After a rod enters transition or film boiling, the GEXL correlation is only used indirectly as described below. To return to nucleate boiling, the following 3 criteria apply:

1. [[

]] [BSEP MELLLA+ SRXB-RAI-9b (Reference 1)]

3.0 Application of PRIME to ODYN and TRACG

The GEH transient analysis model, ODYN, had been used with and is qualified to apply to different fuels. The same method for GNF fuel application is applied for ATRIUM fuel. ATRIUM 10 fuel has been used with ODYN for LaSalle, Columbia, River Bend, and Grand Gulf. ATRIUM 10 ATWS calculations have also been performed using ODYN to support Susquehanna. ATRIUM 10 and ATRIUM 10XM are very similar except for the inventory of full and part-length rods. For the BFN ATWS analyses, the ATRIUM 10XM fuel geometry was modeled explicitly. The core parameters were introduced into ODYN in the same manner via PANAC wrapups as would be done for GNF and ATRIUM 10 fuel designs. ODYN is suitable for ATRIUM 10, and as a result remains suitable for ATRIUM 10XM. [BSEP MELLLA+ SNPB-RAI-7a (Reference 1)]

The PRIME computer model provides best-estimate predictions of the thermal-mechanical performance of nuclear fuel rods. For BFN ATWS analyses, the PRIME code was not used for any thermal-mechanical evaluation. The PRIME code was only used to create the fuel files needed to determine the gap conductance.

Any differences in heat transfer from differences in the cladding thermal properties are insignificant compared to the gap conductance heat transfer. The gap conductivity sensitivities that were performed cover a large range of uncertainty and cover any variation in cladding heat transfer properties. The effect of the large variability in gap conductance had a small effect on PCT (see M+SAR Table 9-11).

The PRIME code, without modification, was only used to create the fuel files needed to determine the gap conductance; therefore, many of the allowable design ranges in PRIME are not applicable. The fuel parameter sensitivity study accounts for uncertainty in the fuel design. [BSEP MELLLA+ SNPB-RAI-7b (Reference 1)]

4.0 TRACG Analysis

4.1. Parameter Selection for Fuel Parameter Sensitivity

The process used to determine the parameters in the ATRIUM 10XM fuel parameters sensitivity study involved creating the Phenomena Identification and Ranking Table (PIRT) that is used in the Code, Scaling, Applicability and Uncertainty (CSAU) methodology. First, the PIRT was developed by GEH experts to identify and rank all phenomena that govern ATWS responses. From these parameters, the highly ranked fuel-related parameters were down selected. From these, each parameter was screened to identify the fuel parameters that have an effect on ATWS application results. Any parameter that (1) did not have appreciable effect, or (2) are independent of the fuel type, were not considered further. The final list was examined by an expert review committee.

The sensitivity ranges were determined as follows:

- [[

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[BSEP MELLLA+ SRXB-RAI-6a (Reference 1)]

5.0 TRACG Results

5.1. T_{min} Sensitivity

Table 9-10 of the M+SAR includes the sensitivities with the lower T_{min}. The difference in PCT between the limiting fuel parameter values and nominal fuel parameter values are larger for cases with the lower T_{min}. When the TRACG homogeneous nucleation T_{min} model is used together with the nominal feedwater temperature reduction rate, the limiting fuel parameter ΔPCT is increased. The main reason for the larger ΔPCT is because T_{min} was exceeded when using the homogeneous nucleation T_{min}. After T_{min} is exceeded, the hot rod surface does not return to nucleate boiling during the flow oscillations and surface temperatures rapidly increase to high temperatures. The same values of the fuel-parameter uncertainties applied to the cases with temperatures above T_{min} are expected to cause a larger effect than the cases with temperatures below T_{min}. For example, [[

]]

[BSEP MELLLA+ SRXB-RAI-7b (Reference 1)]

5.2. Plots

Figure 9-2 in the M+SAR shows the core key parameters from ATWSI RPT case results at MOC with regional mode oscillations. The local fuel assembly behaviors are plotted for the limiting PCT assembly (CHAN115) and its symmetric assembly CHAN116 (See Figures 3 and 4). The plots include time-dependent assembly power, assembly inlet flow rate, and maximum cladding temperature for each of these two assemblies. [[

]] [BSEP MELLLA+ SRXB-RAI-5b (Reference 1)]

6.0 References

1. Email, Andrew Hon (NRC) to William R. Murray (Duke Energy), "Brunswick Unit 1 and Unit 2 Request for Additional Information related MELLLA+ LAR (CACs MF8864 and MF8865) (Proprietary)," January 5, 2018. (ADAMS Accession Numbers ML18010A050 and ML18010A051)
2. GE Hitachi Nuclear Energy, "DSS-CD TRACG Application," NEDE-33147P-A, Revision 4, August 2013.
3. GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.
4. GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," NEDE-32906P-A, Revision 3, September 2006.

[[

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Figure 1. MOC ATWSI TRACG Channel Grouping (Regional Mode)

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Figure 2. MOC ATWSI TRACG Channel Grouping (Core-Wide Mode)

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Figure 3. MOC RPT ATWSI TRACG Assembly Response (Regional Mode)

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Figure 4. MOC RPT ATWSI TRACG Assembly Response (Regional Mode) – Focused Near PCT

ENCLOSURE 3

Framatome Additional Information for Browns Ferry MELLLA+ (Proprietary)

ENCLOSURE 4

Framatome Additional Information for Browns Ferry MELLLA+
(Non-proprietary)

framatome

Framatome Additional Information for Browns Ferry MELLLA+

ANP-3656NP
Revision 1

February 2018

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	1.0	Corrected spelling error “Amendment”

Contents

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Nomenclature

Acronym	Definition
AOO	Anticipated Operational Occurrence
ATWS-I	Anticipated Transient Without Scram Instability
BFN	Browns Ferry Nuclear Plant
BWR	Boiling Water Reactor
CCFL	Counter Current Flow Limit
CHF	Critical Heat Flux
CPR	Critical Power Ratio
DIR	Design Input Request
ECPR	Estimated Critical Power Ratio
EPFOD	Extended Power / Flow Operating Domain
LAR	Licensing Amendment Request
LHGR	Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
NRC	Nuclear Regulatory Commission
OLMCPR	Operating Limit Minimum Critical Power Ratio
PLFR	Partial Length Fuel Rod
SLMCPR	Safety Limit Minimum Critical Power Ratio

1.0 INTRODUCTION

In support of another customer's License Amendment Request (LAR) for an expansion of the core power flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)), the US Nuclear Regulatory Commission (NRC) requested additional information to facilitate the review. Reference 1 presents the requested information.

This document presents information responsive to the applicable requests for additional information for the Browns Ferry Nuclear Plant (BFN) which supports the BFN MELLLA+ LAR.

2.0 PARAMETER SELECTION FOR FUEL PARAMETER SENSITIVITY

The parameters to be investigated were provided to Framatome Inc.* from GEH through a Design Input Request (DIR). This request provided the parameters for which bounding ranges were to be determined. In some cases requests for minimum and maximum values were made, and in others a suggested range was given. As part of this process, five parameters were requested for further study.

Gap Conductance: Minimum and maximum gap conductance multipliers were requested. These multipliers would then be applied to an ATRIUM-10 based gap conductance, to ensure that the gap conductance used in the analysis appropriately bounds the expected ATRIUM 10XM gap conductance. The multipliers were determined by assessing multiple rod histories over a wide range of LHGRs and rod exposures. Each rod history was assessed for both ATRIUM-10 and ATRIUM 10XM. Once all of these histories were calculated, a minimum and maximum were obtained to bound all of the calculated multipliers. In order to prevent unnecessarily conservative results, the multipliers were given as a function of LHGR, and rod exposure.

Void History Bias and Uncertainty Model: Framatome has not identified any bias related to the void history. Specifically, Framatome methodology does not apply uncertainties or biases to any of the reactivity coefficients that are used in the transient analysis.

Direct Moderator Heating: Similar to the gap conductance, the direct moderator heating of the ATRIUM 10XM was compared back to the ATRIUM-10 direct moderator heating and a minimum and maximum adjustment was provided. The evaluation covered a wide range of exposures (0-40 GWd/MTU) and void fractions (0-99%). A comparison of the direct moderator heating fractions between the ATRIUM-10 and ATRIUM 10XM fuel designs shows that the direct

* Framatome Inc. formerly known as AREVA Inc.

moderator energy deposition fraction for ATRIUM 10XM fuel is 3% - 11% lower than the fraction for ATRIUM-10 fuel in the bottom lattice (fully rodded region) and from 2% higher to 6% lower in the top lattice (above the part length rods). The minimum and maximum adjustments were chosen to bound all exposures and void fractions.

Pressure loss: The Framatome pressure drop methodology is described in XN-NF-79-59(P)(A), Methodology for Calculation of Pressure Drop in boiling water reactor (BWR) Fuel Assemblies. As part of the qualification of a new fuel assembly, single phase pressure drop measurements are used to determine component loss coefficients. Further validation of the loss coefficients is performed by comparisons to two phase data. Tests have been performed for an ATRIUM 10XM fuel assembly. This includes two phase pressure drop measurements performed at the KATHY test facility. XCOBRA calculations were performed and benchmarked to these KATHY measurements. The results of this benchmarking show that the ATRIUM 10XM pressure drop prediction is within the statistics given in Table 4.4 of XN-NF-79-59(P)(A) which lists a standard deviation of **[ ]**. A range of +/-10% was selected, of which +/- 5% accounts for the actual uncertainty in the pressure loss data, to conservatively bound the standard deviation in the XCOBRA calculation for ATRIUM 10XM. The range is applicable to both channel losses and leakage losses.

Critical power: As discussed in Section 4.0, the GEXL correlation coefficients were set such that a conservative prediction of minimum critical power ratio (MCPR) is obtained relative to the actual ATRIUM 10XM performance using the same database used to derive the coefficients for the ACE/ATRIUM 10XM correlation. For the purpose of sensitivity studies, a range of critical power prediction was provided to bound the uncertainty in the benchmarked data. A full description of sensitivity range derivation can be found in Section 4.0.

(Reference 1, SRXB-RAI-6a)

3.0 VOID-QUALITY CORRELATIONS

Framatome's independent validation of the BWR void fraction correlations against the FRIGG experiments was performed to supplement and confirm the validation performed by the original developers of the correlations. For each of the experimental tests, the nominal test conditions were used to calculate the void fraction at each of the measurement locations. These calculated void fractions were then compared to the nominal measured void fractions and plotted in Figures A-1 and A-3 in Appendix A. The FRIGG benchmarks provide a basis for assessing the continued applicability of the licensed void fraction correlations to both modern fuel designs and operation of those fuel designs to extended power/flow operating domain (EPFOD) conditions.

Concurrent with the Framatome fuel design development that lead to the introduction of partial length fuel rods (PLFRs) and swirl vane spacer designs, fundamental testing was conducted to assess the adequacy of current methodologies in predicting steady-state behavior (critical power, pressure drop and void fraction) as well as dynamic behavior (channel decay ratio and instabilities). Framatome performed void fraction measurements to specifically assess the impact of the ATRIUM-10 fuel design attributes. The void fraction tests were performed at the KATHY test facility using a prototypical BWR critical heat flux (CHF) test assembly. The test assembly used 8 PLFRs, mixing vane grids, a **[** **]** typical of CHF tests. The tests were conducted using a scanning gamma densitometer that traversed across the test section. Void measurements were made at one of three different elevations in the assembly for each test point: just before the end of the part length fuel rods, midway between the last two spacers, and just before the last spacer.

To assess the high void fraction performance associated with EPFOD, Framatome performed gamma tomography measurements on the ATRIUM 10XM (the fuel design used in Browns Ferry). Again, the tests were performed at the KATHY test facility using a prototypical BWR CHF test assembly. The test assembly used the PLFR and grid

spacer configuration of the ATRIUM 10XM and a **[**

] in the test facility.

Plots presenting the comparisons between the void-quality correlations and measured data were presented in Figures A-2 and A-4 in Appendix A for the steady-state core simulator and transient methods. Visual examinations of these plots as well as comparisons of the mean and standard deviations in Table 1 demonstrate that the modern Framatome fuel design attributes and the potential operation at high void fractions for EPFOD result in no significant change in the void-quality correlation uncertainties relative to the historic validation.

(Reference 1, SNPB-RAI-3a)

The analyses in Sections A.2 and A.3 present the operating limit minimum critical power ratio (OLMCPR) impact of biasing the nominal void fraction correlations over the uncertainty range relative to the inherent conservatism of the approved licensing methodologies on the OLMCPR. Three cases are presented relative to the nominal licensing application:

Case 1: Adjustment of the steady-state simulator void fraction correlation to minimize the mean error over the void fraction range. The resultant steady-state correlation comparison to measured data is presented in Figure A-5. **[**

]

Case 2: Adjustment of Case 1 **[** **]** that would correspond to a **[** **]** offset for the ATRIUM-10 benchmark comparisons. For the steady-state simulator the nodal void fractions calculated from the modified correlation in Case 1 are adjusted by **[**

]

[

]

Case 3: Adjustment of Case 1 **[** **]** that would correspond to a **[** **]** offset for the ATRIUM-10 benchmark comparisons. For the steady-state simulator the nodal void fractions calculated from the modified correlation in Case 1 are adjusted by **[**

]

For each of the cases identified above, a multi-cycle depletion with MICROBURN-B2 was performed to establish the end-of-cycle core conditions (exposure, power flow and reactivity parameters) in equilibrium with the revised void quality correlation. The end-of-cycle condition was then used in the downstream transient analyses to quantify the impact on the OLMCPR.

(Reference 1, SNPB-RAI-3b)

Table 1: Summary of Void-Quality Correlation Statistics

	FRIGG	ATRIUM-10 (KATHY)	ATRIUM 10XM (KATHY)
Core Simulator Mean	[ ]	[ ]	[ ]
Core Simulator Standard Deviation	[ ]	[ ]	[ ]
Transient Method Mean	[ ]	[ ]	[ ]
Transient Method Standard Deviation	[ ]	[ ]	[ ]

4.0 USE OF THE GEXL CORRELATION

The GEXL97 correlation for ATRIUM-10 fuel was transmitted by GEH to Framatome. This transmittal included both the correlation form and the coefficients for ATRIUM-10 fuel. Framatome then modified the coefficients as necessary to conservatively model the ATRIUM 10XM fuel. The data used to benchmark this correlation is the same data set used to benchmark the ACE/ATRIUM 10XM correlation. No changes to the correlation form were permitted, so only changes to the correlation coefficients were allowed. The range of applicability for this correlation was chosen to be within the range of the data available.

(Reference 1, SNPB-RAI-4a)

The GEXL correlation for ATRIUM 10XM was not constructed as a general use correlation. Instead, it was biased to provide a conservative critical power ratio (CPR) prediction specifically for the ATWS-I event. As such, this correlation is not valid for all applications. The correlation was designed to provide conservative estimates for critical power during ATWS-I scenario. This correlation was created to be closer to best estimate, but still conservatively low, at the low flow conditions of interest for the ATWS-I scenario. For other assembly flows the correlation is more conservative. To illustrate this trend, the estimated critical power ratio (ECPR) versus mass flux for the benchmarking data set is shown in Figure 1. This figure shows the trend of increasing conservatism with increasing flow, and also shows the overall conservatism of the correlation.

Figure 1 ECPR Versus Mass Flux for ATRIUM 10XM GEXL Correlation

In order to assess the sensitivity of the critical power calculation, a range of critical power adjustments was provided to conservatively bound the benchmarking uncertainty. Since the correlation itself is not a best estimate correlation where ECPR is approximately 1.0, the uncertainty ranges were adjusted to account for this. Ranges were calculated separately for both the high flow range and the low flow range. As an example, the mean of the low flow ECPR data was found to be **[ ]** with a standard deviation of **[ ]**. Using a **[ ]** sigma multiplier, the adjusted high end ECPR was found to be ECPR = **[ ]**. In order to bring this ECPR back to the best estimate value of 1.0, a **[ ]** decrease in the critical power would be required. For purposes of sensitivity study, a **[ ]** conservative adjustment in critical power was recommended. For the reduced conservatism case, the adjusted ECPR was found to be ECPR = **[ ]**. For the purposes of sensitivity study, a **[ ]** increase (less conservative direction) in critical power was recommended.

The correlation coefficients, a general description of the conservatism applied, the range of applicability, and the range of CPR sensitivity to be examined were transferred from Framatome to GEH.

(Reference 1, SNPB-RAI-4b)

Framatome utilized the ACE K-factor in place of the GEXL R-Factor in the CPR calculation. The subsequent benchmarking to the data described above showed this to be a reasonable assumption. The assembly specific R-factors were calculated by Framatome for all assemblies at each core exposure analyzed for ATWS-I using the K-factor method. In order to ensure that the K-factors indeed produced conservative results with the GEXL correlation, the GEXL correlation was implemented in MICROBURN-B2. Critical power calculations were performed for the limiting assemblies at two exposures (beginning of cycle and end of cycle) to demonstrate that the K-factors and the provided GEXL correlation produced a conservative prediction of critical power when compared to the approved ACE/ATRIUM 10XM correlation. These calculations were performed at both high flow and low flow statepoints and confirmed the overall conservatism of the correlation, as well as the trend in increasing conservatism with increasing flow.

(Reference 1, SNPB-RAI-4d)

5.0 REFERENCES

1. Email, A. Hon (NRC) to B. Murray (Duke Energy), "Brunswick Unit 1 and Unit 2 Request for Additional Information related MELLLA+ LAR (CACs MF8864 and MF8865) (Proprietary)," January 5, 2018. (Accession Numbers ML18010A050 and ML18010A051)
2. N. Zuber and J. A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems," J. Heat Transfer, 1965.
3. P. Coddington and R. Macian, "A Study of the Performance of Void Fraction Correlations Used In the Context of Drift-Flux Two-Phase Flow Models," Nuclear Engineering and Design, 215, 199-216, June 2002.
4. **[ ]**
5. K. Ohkawa and R. T. Lahey, Jr., "The Analysis of CCFL Using Drift-Flux Models," Nuclear Engineering and Design, 61, 1980.
6. S. Misu et al., "The Comprehensive Methodology for Challenging BWR Fuel Assembly and Core Design used at FANP," proceedings on CD-ROM, PHYSOR 2002, Seoul, Korea, October 7-10, 2002
7. XN-NF-84-105(P)(A) Volume 1 Supplement 4, *XCOBRA-T: A Computer Code For BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data*, Advanced Nuclear Fuels Corporation, June 1988.
8. O. Nylund et al., "Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Non-Uniform Radial Heat Flux Distribution," FRIGG-3, R-494/RL-1154, November 1969.
9. J. Skaug et al., "FT-36b, Results of void Measurements," FRIGG-PM-15, May 1968.
10. O. Nylund et al., "Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution," FRIGG-2, R-447/RTL-1007, May 1968.

Appendix A Void-Quality Correlations

A.1 VOID QUALITY CORRELATIONS

The Zuber-Findlay drift flux model (Reference 2) is utilized in the Framatome nuclear and safety analysis methods for predicting vapor void fraction in the BWR system. The model has a generalized form that may be applied to two phase flow by defining an appropriate correlation for the void concentration parameter, Co, and the drift flux, Vgj. The model parameters account for the radially non-uniform distribution of velocity and density and the local relative velocity between the phases, respectively. This model has received broad acceptance in the nuclear industry and has been successfully applied to a host of different applications, geometries, and fluid conditions through the application of different parameter correlations (Reference 3).

Two different correlations are utilized at Framatome to describe the drift flux parameters for the analysis of a BWR core. The correlations and treatment of uncertainties are as follows:

- The nuclear design, frequency domain stability, nuclear anticipated operational occurrence (AOO) transient and accident analysis methods use the **[** **]** void correlation (Reference 4) to predict nuclear parameters. Uncertainties are addressed at the overall methodology and application level rather than individually for the individual correlations of each method. The overall uncertainties are determined statistically by comparing predictions using the methods against measured operating data for the reactors operating throughout the world.

- The thermal-hydraulic design, system AOO transient and accident analysis, and loss of coolant accident (only at specified junctions) methods use the Ohkawa-Lahey void correlation (Reference 5). This correlation is not used in the direct computation of nuclear parameters in any of the methods. Uncertainties are

addressed at the overall methodology level through the use of conservative assumptions and biases to assure uncertainties are bounded.

The **[ ]** void correlation was developed for application to multi-rod geometries operating at typical BWR operating conditions using multi-rod data and was also validated against simple geometry data available in the public domain. The correlation was defined to be functionally dependent on the mass flux, hydraulic diameter, quality, and fluid properties.

The multi-rod database used in the **[**

]. As a result, the multi-rod database and prediction uncertainties are not available to Framatome. However, the correlation has been independently validated by Framatome against public domain multi-rod data and proprietary data collected for prototypical ATRIUM-10 and ATRIUM 10XM test assemblies. Selected results for the ATRIUM-10 test assembly are reported in the public domain in Reference 6.

The Ohkawa-Lahey void correlation was developed for application in BWR transient calculations. In particular, the correlation was carefully designed to predict the onset of counter current flow limit (CCFL) characteristics during the occurrence of a sudden inlet flow blockage. The correlation was defined to be functionally dependent on the mass flux, quality, and fluid properties.

Independent validation of the Ohkawa-Lahey correlation was performed by Framatome at the request of the NRC during the NRC review of the XCOBRA-T code. The NRC staff subsequently reviewed and approved Reference 7, which compared the code to a selected test from the FRIGG experiments (Reference 8). More recently the correlation has been independently validated by Framatome against additional public domain multi-rod data and proprietary data collected for prototypic ATRIUM-10 and ATRIUM 10XM test assemblies, as described below.

The characteristics of the Framatome multi-rod void fraction validation database are listed in Table A-1.

The FRIGG experiments have been included in the validating database because of the broad industry use of these experiments in benchmarking activities, including TRAC, RETRAN, and S-RELAP5. The experiments include a wide range of pressure, subcooling, and quality from which to validate the general applicability of a void correlation. However, the experiments do not contain features found in modern rod bundles such as part length fuel rods and mixing vane grids. The lack of such features makes the data less useful in validating correlations for modern fuel designs. Also the reported instrument uncertainty for these tests is provided in Table A-1 based on mockup testing. However, the total uncertainty of the measurements (including power and flow uncertainties) is larger than the indicated values.

Because of its prototypical geometry, the ATRIUM-10 and ATRIUM 10XM void data collected at KATHY was useful in validating void correlation performance in modern rod bundles that include part length fuel rods, mixing vane grids, and prototypic axial/radial power distributions. Void measurements were made at one of three different elevations in the bundle for each test point: just before the end of the part length fuel rods, midway between the last two spacers, and just before the last spacer.

As shown in Figure A-7, the range of conditions for the ATRIUM void data are valid for typical reactor conditions. This figure compares the equilibrium quality at the plane of measurement for the ATRIUM 10XM void data with the exit quality of bundles in the EMF-2158 benchmarks and Browns Ferry operating at EPFOD conditions. As seen in the figure, the data at the measurement plane covers nearly the entire range of reactor conditions. However, calculations of the exit quality from the void tests show the overall test conditions actually envelope the reactor conditions.

Figure A-1 and Figure A-2 provide comparisons of predicted versus measured void fractions for the Framatome multi-rod void fraction validation database using the **[ ]** correlation. These figures show the predictions fall within ±0.05 (predicted – measured) error bands with good reliability and with very little bias. Also, there is no observable trend of uncertainty as a function of void fraction.

Figure A-3 and Figure A-4 provide comparisons of predicted versus measured void fractions for the Framatome multi-rod void fraction validation database using the Ohkawa-Lahey correlation. In general, the correlation predicts the void data with a scatter of about ±0.05 (predicted – measured), but a bias in the prediction is evident for voids between 0.5 and 0.8. The observed under prediction is consistent with the observations made in Reference 9.

In conclusion, validation using the Framatome multi-rod void fraction validation database has shown that both drift flux correlations remain valid for modern fuel designs. Furthermore, there is no observable trend of uncertainty as a function of void fraction. This shows there is no increased uncertainty in the prediction of nuclear parameters at EPFOD conditions within the nuclear methods when applied to the Browns Ferry reactor.

A.2 VOID QUALITY CORRELATION UNCERTAINTIES

The Framatome analyses methods and the correlations used by the methods are applicable for modern fuel designs in both pre-EPFOD and EPFOD conditions. The approach for addressing the void-quality correlation bias and uncertainties remains unchanged and is applicable for Browns Ferry operation at EPFOD conditions.

The OLMCPR is determined based on the safety limit MCPR (SLMCPR) methodology and the transient analysis (ΔCPR) methodology. Void-quality correlation uncertainty is not a direct input to either of these methodologies; however, the impact of void-correlation uncertainty is inherently incorporated in both methodologies as discussed below.

The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculation performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void-quality correlation uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analysis methodology is not a statistical methodology and uncertainties are not directly input to the analyses. The transient analyses methodology is a deterministic, bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations.

The transient analyses methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analyses methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

Based on the above discussions, the impact of void-quality correlation uncertainty is inherently incorporated in the analytical methods used to determine the OLMCPR. Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void-quality correlation uncertainty.

A.3 BIASING OF THE VOID-QUALITY CORRELATION

Framatome has performed studies to determine the OLMCPR sensitivity to biases approaching the upper and lower extremes of the data comparisons shown in Figure A-1 through Figure A-4.

For one of these studies, the transient ΔCPR impact was determined by propagating void-quality biases through three main computer codes: MICROBURN-B2, COTRANSA2, and XCOBRA-T.

The **[ ]** correlation in MICROBURN-B2 was modified to correct the mean to match the measured ATRIUM-10 void fraction data shown in Figure A-2. The modified **[ ]** correlation parameters were then modified to generate two bounding correlations for the ATRIUM-10 of ±0.05 void. The results of this modified correlation are presented in Figure A-5.

COTRANSA2 does not have the **[ ]** correlation. For COTRANSA2 the modified **[ ]** correlations in MICROBURN-B2 were approximated in COTRANSA2 with **[ ]**. Figure A-6 shows a comparison of the **[ ]** ratio results compared to the ATRIUM-10 test data. This approach created equivalent void fractions as the **[ ]** correlation modifications.

The thermal hydraulic methodology incorporates the effects of subcooled boiling through use of the Levy model. The Levy model predicts a critical subcooling that defines the onset of boiling. The critical subcooling is used with a profile fit model to determine the total flow quality that accounts for the presence of subcooled boiling. The total flow quality is used with the void-quality correlation to determine the void fraction. This void fraction explicitly includes the effects of subcooled boiling. Application of the Levy model results in a continuous void fraction distribution at the boiling boundary.

Like COTRANSA2, XCOBRA-T does not have the **[** **]** correlation. Unlike COTRANSA2, XCOBRA-T does not have **[** **]**. For the other void scenarios, no correction was done in XCOBRA-T. Not modifying the void-quality correlation for the other void scenarios results in a very small difference in ΔCPR.

The transient response was assessed relative to a limiting uprated BWR plant transient calculation. The impact of the change in the void correlations was also captured in the burn history of the fuel (the results are not for an instantaneous change in the void correlations). The SLMCPR response was also assessed with the new input corresponding to the three different void scenarios. The results are provided in Table A-2.

The major influence that the void-quality models have on scram reactivity worth is through the predicted axial power shape. The void-quality models, used for ATRIUM fuel, result in a very good prediction of the axial power shape.

As seen in the results in Table A-2, modifying the void-quality correlations to correct the mean to match the measured ATRIUM-10 void fraction data results in a very small increase in ΔCPR, a very small decrease in SLMCPR, and a very small increase in OLMCPR for this study; therefore, the impact of the correlation bias is insignificant.

The +0.05 void scenarios show an increase in the OLMCPR; however, as mentioned previously, the transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the models to bound results on an integral basis relative to benchmark tests. For licensing calculations, important input parameters are biased in a conservative direction. In addition, the licensing calculations include a 110% multiplier to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analysis methodology (which includes the void-quality correlation). Even with an extreme bias in the void correlation of +0.05, the

conservatism introduced by the 110% multiplier is alone sufficient to offset the increase in results presented in Table A-2. For the study, the conservatism of the 110% multiplier was **[ ]**. These calculations demonstrate that the overall methodology has sufficient conservatism to account for both the bias and the uncertainty in the void-quality correlation.

To provide a more accurate assessment of the impact of a +0.05 void bias, Framatome would need to re-evaluate the Peach Bottom transient benchmarks; it is likely that the +0.05 void scenario would show overconservatism in the benchmarks. Likewise, the pressure drop correlations and core monitoring predictions of power will likely show a bias relative to measured data. Correcting the models to new benchmarks and measured data would further reduce the OLMCPR sensitivity.

A.4 VOID-QUALITY CORRELATION UNCERTAINTY SUMMARY

Integral power is a parameter obtainable from test measurements that is directly related to ΔCPR and provides a means to assess code uncertainty. The COTRANSA transient analysis methodology was a predecessor to the COTRANSA2 methodology. The integral power figure of merit was introduced with the COTRANSA methodology as a way to assess (not account for) code uncertainty impact on ΔCPR. From COTRANSA analyses of the Peach Bottom turbine trip tests, the mean of the predicted to measured integral power was 99.7% with a standard deviation of 8.1%. Framatome (Exxon Nuclear at the time) initially proposed to treat integral power as a statistical parameter. However, following discussions with the NRC, it was agreed to apply a deterministic 110% integral power multiplier (penalty) on COTRANSA calculations for licensing analyses. That increase was sufficient to make the COTRANSA predicted to measured integral power conservative for all of the Peach Bottom turbine trip tests.

COTRANSA2 is not a statistical methodology and uncertainties are not directly input to the analyses. The methodology is a deterministic bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed

to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations. Justification that the integrated effect of all the conservatisms in COTRANSA2 licensing analyses is adequate for EPFOD operation at Browns Ferry is provided below.

The COTRANSA2 methodology results in predicted power increases that are bounding (**[ ]** on average) relative to Peach Bottom benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism. This approach adds significant conservatism to the calculated OLMCPR as discussed previously.

Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. The Peach Bottom turbine trips were performed assuming the measured performance of important input parameters such as control rod scram speed and turbine valve closing times. For licensing calculations, these (and other) parameters are biased in a conservative bounding direction. These conservative assumptions are not combined statistically; assuming all parameters are bounding at the same time produces very conservative results.

With the ATRIUM 10XM void fraction benchmarks presented in Figure A-2 and Figure A-4, the applicability of the void-quality correlation at high void fractions is confirmed and the uncertainty associated with the application of the correlation to the ATRIUM 10XM design is demonstrated to be equivalent to the data used in the bias assessment. Therefore, the sensitivity studies and conclusions drawn from the study are equally applicable to EPFOD operation of the ATRIUM 10XM at Browns Ferry.

Table A-1 Framatome Multi-Rod Void Fraction Validation Database

	FRIGG-2 (Reference 10)	**FRIGG-3 (Reference 8 & 9)**	**ATRIUM-10 KATHY**	**ATRIUM 10XM KATHY**
Axial Power Shape	**uniform**	**uniform**	**[ ]**	**[ ]**
Radial Power Peaking	uniform	mild peaking	**[ ]**	**[ ]**
Bundle Design	circular array with 36 rods + central thimble	circular array with 36 rods + central thimble	**[ ]**	**[ ]**
Pressure (psi)	725	725, 1000, and 1260	**[ ]**	**[ ]**
Inlet Subcooling (^{0}F)	4.3 to 40.3	4.1 to 54.7	**[ ]**	**[ ]**
Mass Flow Rate (lbm/s) *(Based on mass flux assuming ATRIUM-10 inlet area)*	14.3 to 31.0	10.1 to 42.5	**[ ]**	**[ ]**
Equilibrium Quality at Measurement Plane (fraction)	-0.036 to 0.203	-0.058 to 0.330	**[ ]**	**[ ]**
Max Void at Measurement Plane (fraction)	0.828	0.848	**[ ]**	**[ ]**
Reported Instrument Uncertainty (fraction)	0.025	0.016	**[ ]**	**[ ]**
Number of Data	27 tests, 174 points	39 tests, 157 points	**[ ]**	**[ ]**

Table A-2 Void Sensitivity Results

Parameter	Reference Calculation	Modified V-Q (0.0)	Modified V-Q (+0.05)	Modified V-Q (−0.05)
ΔCPR	0.305	0.307	0.321	0.305
SLMCPR	1.09	1.09	1.09	1.09
ΔSLMCPR	NA	−0.001	−0.002	+0.002
OLMCPR	1.395	1.396	1.409	1.397

Figure A-1 Validation of [ ] using FRIGG-2 and FRIGG-3 Void Data

Figure A-2 Validation of [ ] using ATRIUM-10 and ATRIUM 10XM Void Data

Figure A-3 Validation of Ohkawa-Lahey using FRIGG-2 and FRIGG-3 Void Data

Figure A-4 Validation of Ohkawa-Lahey using ATRIUM-10 and ATRIUM 10XM Void Data

[ ]

Figure A-5 Modified Void Fraction Correlation Comparison to ATRIUM-10 Test Data

[ ]

Figure A-6 [ ] Void Fraction Comparison to ATRIUM-10 Test Data

Figure A-7 Comparison of KATHY Two-Phase Pressure Drop and Void Fraction Test Matrices and Typical Browns Ferry Reactor Conditions with EPFOD

ENCLOSURE 5

Affidavit – GE-Hitachi Nuclear Energy Americas LLC

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Lisa K. Schichlein, state as follows:

(1) I am a Senior Project Manager, NPP/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report 004N6892-P Revision 1, "Browns Ferry Nuclear Plant Units 1, 2, and 3 MELLLA+ ATWS Supplemental Information," dated February 2018. GEH proprietary information in 004N6892-P Revision 1 is identified by a dotted underline within double square brackets. [[This sentence is an example.{3}]]. GEH proprietary information in figures and large objects is identified by double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. §552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. §1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without a license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions for proprietary or confidentiality agreements or both that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of the Maximum Extended Load Line Limit Analysis Plus analysis for a GEH Boiling Water Reactor ("BWR"). The analysis utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of Maximum Extended Load Line Limit Analysis Plus for a GEH BWR.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and

technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

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

Executed on this 20th day of February 2018.

Lisa K. Schichlein

Lisa K. Schichlein
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ENCLOSURE 6

Affidavit – Framatome Inc.

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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-3656P, Revision 1, "Framatome Additional Information for Browns Ferry MELLLA+," dated February 2018 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.

SUBSCRIBED before me this 26th

day of February, 2018.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON