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

CNL-16-116

July 27, 2016

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-505 - Request for License Amendments - Extended Power Uprate (EPU) - Supplement 25, Responses to Requests for Additional Information**

References:
1. Letter from TVA to NRC, CNL-15-169, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU)," dated September 21, 2015 (ML15282A152)
2. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC Nos. MF6741, MF6742, and MF6743)," dated June 21, 2016 (ML16154A544)

By the Reference 1 letter, Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for the Extended Power Uprate (EPU) of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed LAR modifies the renewed operating licenses to increase the maximum authorized core thermal power level from the current licensed thermal power of 3458 megawatts to 3952 megawatts. During their technical review of the LAR, the Nuclear Regulatory Commission (NRC) identified the need for additional information. The Reference 2 letter provided NRC Requests for Additional Information (RAIs) related to spent fuel pool criticality safety analysis. The due date, provided by the Reference 2 letter, for the responses to these NRC RAIs is July 29, 2016.

Enclosure 1 of this letter provides the responses to NRC RAIs ESGB-RAI 2, ESGB-RAI 3, SFP-RAI 5, SFP-RAI 11, and SFP-RAI 12 from the Reference 2 letter.

Enclosure 2 of this letter provides the response to the remaining NRC RAIs from the Reference 2 letter. AREVA considers portions of the information provided in Enclosure 2 to this letter to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390, Public inspections, exemptions, requests for withholding. An affidavit for withholding information, executed by AREVA, is provided in Enclosure 4. A non-proprietary version of the RAIs and responses are provided in Enclosure 3. Therefore, on behalf of AREVA, TVA requests that Enclosure 2 be withheld from public disclosure in accordance with the associated AREVA 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 1 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 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 Mr. Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th day of July 2016.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures

cc: See Page 3

Enclosures:

1. Responses to NRC Requests for Additional Information ESGB-RAI 2, ESGB-RAI 3, SFP-RAI 5, SFP-RAI 11, and SFP-RAI 12
2. ANP-3495P, Revision 1, Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2 (Proprietary version)
3. ANP-3495NP, Revision 1, Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2 (Non-proprietary version)
4. AREVA Affidavit

cc:

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama Department of Public Health (w/o Enclosure 2)

ENCLOSURE 1

Responses to NRC Requests for Additional Information
ESGB-RAI 2, ESGB-RAI 3, SFP-RAI 5, SFP-RAI 11, and SFP-RAI 12

ESGB-RAI 2

In NUREG-1801, Revision 2 (published December 2010) Generic Aging Lessons Learned Report, Aging Management Program XI.M40, the staff has established a 10-year maximum surveillance interval for the inspection and testing of a neutron absorbing material other than Boraflex in the SFP. This report also specifies that the periodic testing should include measurement of areal density via coupons or in situ techniques. The program described in your RAI response, dated February 16, 2016, is consistent with the staff's expected time interval for testing of your coupons, however you state that neutron attenuation testing to verify Boron-10 (B-10) areal density is not included in the program. You further state that neutron attenuation testing has not been performed on the coupons at BFN since 1985. Please describe your plans to perform neutron attenuation testing in the future. Include a discussion of any modifications to your current program to require attenuation testing.

TVA Response:

TVA will perform areal density measurements on one Boral sample prior to Extended Power Uprate (EPU) implementation at the Browns Ferry Nuclear Plant (BFN).

In addition, as part of the EPU License Amendments, either BFN will accept license conditions for the performance of periodic Boral areal density measurement worded as follows.

> "The licensee shall, at least once every ten years, withdraw a neutron absorber coupon from the spent fuel pool and perform Boron-10 (B-10) areal density measurement on the coupon. Based on the results of the B-10 areal density measurement, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes."

However, if NRC endorses NEI 16-03, "Guidance on Neutron Absorber Monitoring," on neutron absorber monitoring prior to issuance of the EPU license amendments, BFN will accept license conditions for the performance of periodic Boral areal density measurement worded as follows.

> "The licensee shall perform tests in accordance with NRC-endorsed NEI 16-03, "Guidance on Neutron Absorber Monitoring." Based on the results of the testing performed in accordance with NEI 16-03, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes."

Upon issuance of the EPU License Amendments, the current Boral monitoring program will be modified accordingly.

ESGB-RAI 3

The Boral coupon surveillance program at BFN relies on coupons in the Unit 3 SFP to represent the Boral in all three units. Describe why the Unit 3 coupons are representative or bounding of the service conditions for the Units 1 and 2 Boral. Please discuss the cumulative time that the actual Boral panels have been in service in each unit and compare that time to the Unit 3 coupons. The response should address similarities and differences in the temperature, flow, water chemistry, and irradiation levels between the SFPs as well as similarities and differences in the materials and manufacturing processes of the Boral panels between the SFPs.

TVA Response:

The BFN High Density Spent Fuel Racks (HDSFRs) were supplied by General Electric Uranium Management Company (GEUMCO), a subsidiary of General Electric. All of the Boral plates and coupons were provided by a single supplier, namely Brooks & Perkins. Based on GEUMCO Product Quality Certification records, all of the racks were certified prior to the end of 1983. The coupons were installed in the Unit 3 SFP in late 1983. Therefore, the materials and methods of construction for the Boral plates and the coupons can be considered similar, if not identical.

The first HDSFRs were installed in BFN Unit 3 Spent Fuel Pool (SFP) in 1978. The remainder (57 total) of the HDFSRs were installed in phases from that time until 1999. The Boral coupons were first installed in the BFN Unit 3 SFP in 1983; therefore, the coupon installation postdated the first HDFSR installation. However, the coupons have resided in the SFP continuously for 33 years and, as reported in the TVA response to ESGB-RAI 1, have been removed and inspected on multiple occasions since their installation. The three BFN SFPs contain a total of 20,940 Boral panels. A total of 1464 panels pre-date the coupons by approximately five years. Another 1908 panels pre-date the coupons by approximately four years. The full history of rack installation relative to coupons installation is provided in the table below.

Summary of Panel Lag / Lead vs Coupons

Dates Modules Installed	Unit	Time Lag / Lead (yrs)	No. of Panels Affected	% of Panels Affected
08/10/1978	3	-5.18	1464	6.99
07/22/1979	1	-4.23	1908	9.11
08/09/1980	2	-3.18	1908	9.11
10/26/1981	3	-1.97	444	2.12
04/25/1983	1	-0.47	5072	24.22
09/02/1983	3	-0.12	340	1.62
10/14/1983	3	Coupons Installed in BFN Unit 3 SFP		
09/12/1984	2	0.92	2352	11.23
03/01/1986	2	2.38	2040	9.74
08/01/1986	3	2.80	2484	11.86
01/01/1999	2 and 3	15.23	2928	13.98
Total			20940	100

Note: (-) means panels lag the coupons

The BFN SFP water chemistry and temperature are maintained within the limits specified in the BFN Technical Requirements Manual (TRM) Sections 3.9.2 and 3.9.3. The TRM requirements for all three units are as follows.

- TRM 3.9.2 – Fuel pool water temperature shall be < 150°F.
- TRM 3.9.3 – Fuel pool water shall be maintained within the following limits:
  - Conductivity < 10 µmhos/cm at 25°C
  - Chlorides < 0.5 ppm.

A review of chemistry data taken as far back as 2000 indicated that SFP conductivity remained at approximately 1 µmhos/cm for the majority of the time and only very rarely exceeded 2 µmhos/cm for all three SFPs. During the same period, the BFN SFPs also remained significantly below the chloride limit with normal readings under 0.03 ppm and with only occasional excursions approaching 0.08 ppm. SFP temperatures are procedurally controlled to remain between 75°F and 125°F and no information indicated that SFP temperatures approached the 150°F TRM limit.

From the review, it is also reasonable to assume that the BFN Unit 3 SFP environment was similar to the BFN Units 1 or 2 SFP environments. The BFN Unit 3 Boral coupons and the Boral plates shared a common environment. Therefore, the growth and abatement of the corrosion seen on the coupons in BFN Unit 3 would be representative for the HDSFR Boral plates in BFN Units 1 and 2.

Because the Boral plates are contained in a stainless steel sheath, there is little communication with the pool in general and the Boral plates are not susceptible to mechanical disturbance of the corrosion layer that is developed soon after installation. As a result, once the passivation layer is formed, there is little to no change in the Boral corrosion. The coupons, which were installed in the BFN SFPs in 1983, have been removed and inspected 12 times since then, with the first inspection occurring in 1985. The later inspections that were performed, as late as

2010, have shown very little or no change in coupon corrosion growth since the earlier inspections, thus providing further assurance of the stability of the Boral plates.

The Spent Fuel Pool Cooling and Cleanup system design is similar for all three units and maintains cooling and clean-up flow in the pools to within 600 GPM to 1,200 GPM each. Each SFP contains 19 HDSFRs and all are of similar size and shape such that the general flow velocity is similar for all three SFPs. Therefore, the HDSFRs and coupons have been exposed to very similar water flow conditions and no flow induced corrosion issues could be considered credible for the HDSFR Boral plates with respect to the coupons.

A review of the fuel assembly movements since plant start-up was performed to determine the total Core Average Exposure (CAVEX) for each of the three SFPs. The information is tabularized below.

UNIT	ESTIMATED CAVEX FROM HDSFR INSTALLATION (1978) UP TO COUPON INSTALLATION (1983) (MWD/ST)	ESTIMATED CAVEX POST COUPON INSTALLATION (MWD/ST)	ESTIMATED TOTAL CAVEX (MWD/ST)	DELTA TOTAL EXPOSURE FROM UNIT 3 COUPONS (%)
1	23,353	41,561	64,914	-54.4%
2	14,720	170,985	185,705	+23.4%
3	26,569	142,247	168,816	+15.7%

All three SFPs have similar and mild chemistry and temperature histories so the BFN Unit 3 Boral coupons are representative of the Boral plates in all three units when considering long term environmental impacts. The inspections that have been performed on the BFN Unit 3 coupons have shown very little or no change in corrosion and similarities in the SFP environments provide assurance that the Boral plates in the BFN Unit 1 and 2 SFPs remain in acceptable physical condition as well.

However, from an exposure perspective, some rack modules have accumulated slightly more exposure than that of the Boral coupons. As an enhancement to the BFN neutron absorber monitoring program, the coupons will be managed such that their exposure is accelerated.

SFP-RAI 5

ANP-3160 states that the Boral is modeled using the design minimum B-10 areal density. The staff was unable to locate information describing how the Boral installed in the BFN SFP racks was verified to meet this minimum value, including any applicable measurement uncertainties. Provide an explanation of how the minimum B-10 areal density used in the SFP criticality analysis bounds all of the Boral plates installed in the BFN SFP racks.

TVA Response:

In the Reference 1 letter, TVA described the method used to ensure the Boron-10 (B-10) areal density of the Boral plates installed in the BFN SFP racks bounds the minimum areal density specified for the Boral plates (i.e., the minimum B-10 areal density assumed in the SFP criticality safety analysis). Section 3, Materials, of Reference 1 states, in part:

> "To provide assurance that specification Boral* sheet is utilized during tube fabrication, a special quality control program is in effect at the manufacturer's facility. Samples of each Boral sheet are chemically analyzed to determine the B^{10} content. These data are evaluated to verify that the samples are statistically representative of the entire area of the Boral plate and that B^{10} content, at a 95% confidence level, meets of exceeds specification requirements. Analyses are also performed to establish the correlation between the B^{10} content and the thickness of the Boral sample. The Boral sheets are dimensionally inspected and the thickness data are statistically analyzed to verify the sheet meets the minimum thickness requirement over its entire area at a 95% confidence level. These thickness data are also compared with the correlation data to provide additional assurance that the B^{10} content meets or exceeds specification requirements. Before each piece of Boral is inserted into a tube assembly it is verified that each inspection has been successfully performed. Presence of the neutron absorber material in the fabricated fuel storage module is verified by dimensional inspection and by visual examination."
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> "*Product of Brooks & Perkins, Inc. consisting of a layer of B_4C-Al matrix bonded between two layers of aluminum."

Reference

1. Letter from TVA to NRC, "Revised Design Basis And Environmental Assessment for the Proposed High Density Fuel Storage System," dated December 2, 1977 (ADAMS Accession No. 4008006709)

SFP-RAI 11

The response to SFP-RAI 1 (ADAMS Accession No. ML16049A248) discusses requirements given in the criticality safety analysis (CSA) report (AREVA Report ANP-3160, Revision 1, attachment to ADAMS Accession No. ML15351A097) that must be met for fuel stored in the BFN SFPs. Fuel stored in the BFN SFPs may meet either one of two sets of requirements. The first set of requirements impose limits on the U-235 and gadolinia loading for the top and bottom lattices. The second set of requirements represent an in-rack k-infinity limit for the top and bottom lattices. The first set implicitly confirms that the second set is met, since the criticality analysis demonstrates that the maximum U-235 loadings combined with the minimum gadolinia loadings will result in an in-rack k-infinity that meets the provided limits. The NRC staff has determined that some of the margin to the regulatory limit as calculated in the CSA report will need to be credited to address potential nonconservatisms. Therefore, for the NRC staff to make a safety finding, the upper limit of the reactivity of the fuel lattices used in the CSA must be maintained. The Standard Technical Specifications list a limit on the k-infinity as calculated in the standard cold core geometry for fuel stored in SFPs, which is an example of one way of meeting this condition.

Propose a means of ensuring that the maximum reactivity of fuel assemblies loaded in the SFP will be limited such that the NRC safety finding for this CSA will continue to be valid for future fuel assemblies stored in the BFN SFPs.

TVA Response:

Changes to BFN Technical Specification 4.3.1.1 are proposed to provide control of the maximum reactivity of fuel assemblies stored in the SFP. The proposed limit is consistent with the BFN SFP CSA documented in ANP-3160(P), Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM 10XM Fuel, Revision 1. The proposed changes are consistent with those approved for Monticello (reference ML15160A207). The markup and retyped pages for each of the three BFN units are provided in Attachments 1 and 2, respectively of this enclosure.

SFP-RAI 12

In response to SFP-RAI 2, the licensee indicated a review of plant records showed that the Boral verification testing was inconclusive for some cell locations on six BFN SFP storage modules. In order to evaluate these cell locations, a statistical analysis was performed to determine the probability that no more than one Boral plate is missing. The staff notes that if a single Boral plate is missing, then this would become part of the normal condition and should be evaluated as such. Since such a normal condition could result in a more limiting accident condition, provide the following:

a. The locations of the cells for which the Boral verification testing was inconclusive.

b. The statistical analysis performed to evaluate the cell locations for which the Boral verification testing was inconclusive.

c. A technical justification for the assumption that the normal BFN SFP rack condition does not include any missing Boral plates.

d. As part of an audit performed May 10 - 11, 2016, TVA provided draft documentation that included information to address one missing Boral panel as part of the normal condition. In addition to submitting this information on the docket, the NRC staff requests the following information:

i. A discussion of the applicability of the Edwin I. Hatch Nuclear Power Plant (Hatch) testing results to the BFN SFPs, more specifically addressing how the licensee has determined that the manufacturing process used to fabricate the SFP racks delivered to Hatch is the same as those delivered to BFN, and

ii. Information about any similar testing results from other sites that the licensee may be aware of.

TVA Response:

a. During the installation of the High Density Spent Fuel Racks (HDSFRs) at BFN Units 1, 2, and 3, testing was conducted to verify the presence of Boral plates in the racks. The tests were conducted using a neutron source in conjunction with four neutron detectors. The source was placed inside one of the tubes and the neutron signal strength in the four detectors was used to validate the presence (or absence) of the Boral plates.

As noted in the BFN response to SFP-RAI 2, testing in some cell locations in six racks (five in BFN Unit 1 and one in BFN Unit 3) proved inconclusive due to the proximity of irradiated fuel to the detectors (see Table 1 for details). Note that each unverified cell (i.e., Boral tube) location results in four unverified Boral plates because a tube has one Boral plate on each of the four walls. In the presence of gamma radiation, the detectors become “saturated” and their reading drops to zero, resulting in no signal and thus, an indeterminate Boral verification. This event occurred to some cell locations for six racks, as shown below:

ENCLOSURE 1

Table 1, Unverified Boral Plates and Rack Location

Rack Size and Serial Number		Unit/Rack	Unverified Boral Plates	Installed	Tested
13x13	0002	1/4	52	<4/25/1983	6/1983
13x13	0005	1/1	52	<4/25/1983	6/1983
13x17	0021	1/9	148	<4/25/1983	6/1983
13x17	0022	1/17	104	<4/25/1983	6/1983
13x17	0023	1/13	156	<4/25/1983	6/1983
13x17	2336-6*	3/5	136	10/26/1981	10/26/1981

*Rack inspection details are shown in Table 2.

In the five BFN Unit 1 racks, the exact location of the unverified cells is not currently known. TVA is researching original construction and testing information for the racks in an attempt to determine the locations of the unverified cells.

For Rack 2336-6 in BFN Unit 3, the test record (strip chart) identified the following.

- Test results for the following cells were inconclusive as indicated by the strip chart reading 0% for at least one detector and being annotated on the strip chart as being near irradiated fuel.
  L1, L3, L5, L7, L9, L11, L13, L15
  M2, M4, M6, M8, M10, M12, M14, M16
  N1, N3, N5, N7, N9, N11, N13, N15, N17
  K2, K4, K6
- Cell L17 and the entire K row had readings that were approximately 1/2 of the normal readings of cells not affected by adjacent fuel. These cells are being considered inconclusive.

The following table illustrates the unverified cell locations for this particular rack.

Table 2, Unit 3 Rack 2336-6 Unverified Cell Locations

A1		A3		A5		A7		A9		A11		A13		A15		A17
	B2		B4		B6		B8		B10		B12		B14		B16	
C1		C3		C5		C7		C9		C11		C13		C15		C17
	D2		D4		D6		D8		D10		D12		D14		D16	
E1		E3		E5		E7		E9		E11		E13		E15		E17
	F2		F4		F6		F8		F10		F12		F14		F16	
G1		G3		G5		G7		G9		G11		G13		G15		G17
	H2		H4		H6		H8		H10		H12		H14		H16	
J1		J3		J5		J7		J9		J11		J13		J15		J17
	K2		K4		K6		K8		K10		K12		K14		K16	
L1		L3		L5		L7		L9		L11		L13		L15		L17
	M2		M4		M6		M8		M10		M12		M14		M16	
N1		N3		N5		N7		N9		N11		N13		N15		N17

It is anticipated that the unverified cell positions in the BFN Unit 1 racks are located in similar positions, that is, along one or more sides of the racks that bordered other racks with spent fuel, but those locations cannot be verified at this time.

b. Because of this issue of unverified cells, the vendor at that time, GEUMCO, performed a probability analysis to determine the likelihood that there are, in fact, multiple Boral plates missing from the racks. The analysis used a binomial probability function to determine what the probability was for certain numbers of Boral plates to be missing. This type of analysis is typical for manufacturing process errors. It should also be noted that all of the Boral plates and tubes used in fabrication of the racks were provided by the same manufacturer with the same processes and procedures. This is important to note because this further establishes the validity of the methodology and provides greater confidence that all of the Boral plates were properly installed.

In order to perform this type of an analysis, a failure rate had to be established. In this case, a “failure” is defined as a missing Boral plate. The failure rate was established based on prior testing history. Prior to performing the analysis, a total of 36 separate racks that contained 17,304 pieces of Boral had been tested by GEUMCO at Plant Hatch and BFN. In all cases of these inspections, the tubes were shown to contain the required Boral plates, so they were all “successes” in terms of probability. In order to obtain a failure rate for the probability equation, it was assumed that the very next Boral plate was missing. This equates to one failure in 17,305 tries, or a process failure rate of 5.78×10^{-5}. Once the failure rate is established, the following equation describes the probability for missing Boral plates.

$$P(x;p,n) = \binom{n}{x} \times (p^x) \times (1-p)^{(n-x)}\ for\ x = 0,1,2,3 \ldots$$

Where P = probability for a given outcome (in this case, for x missing plates)
x =number of missing plates
n = sample size
p = process failure rate

Also, $\binom{n}{x} = \frac{n!}{x!(n-x)!}$

Setting the sample size equal to the number of unverified Boral plates for each rack, the following table of probabilities can be generated:

Where R = the number of missing Boral plates

For racks with 52 unverified plates:

Number of assumed missing Plates (R)	Probability of the number of missing plates equal to R	Probability of missing plates being equal to or less than R	Probability of missing plates being equal to or greater than R
0	0.996999512	9.96999512E-01	1.000000000000
1	0.002996069	9.99995581E-01	0.003000488211

ENCLOSURE 1

For racks with 104 unverified plates:

Number of missing Plates (R)	Probability of the number of missing plates equal to R	Probability of missing plates being equal to or less than R	Probability of missing plates being equal to or greater than R
0	0.994008027	9.94008027E-01	1.000000000000
1	0.005974158	9.99982185E-01	0.005991973493

For racks with 136 unverified plates:

Number of missing Plates (R)	Probability of the number of missing plates equal to R	Probability of missing plates being equal to or less than R	Probability of missing plates being equal to or greater than R
0	0.992171576	9.92171576E-01	1.000000000000
1	0.007797927	9.99969503E-01	0.007828424392

For racks with 148 unverified plates:

Number of missing Plates (R)	Probability of the number of missing plates equal to R	Probability of missing plates being equal to or less than R	Probability of missing plates being equal to or greater than R
0	0.991483782	9.91483782E-01	1.000000000000
1	0.008480097	9.99963879E-01	0.008516218415

For racks with 156 unverified plates:

Number of missing Plates (R)	Probability of the number of missing plates equal to R	Probability of missing plates being equal to or less than R	Probability of missing plates being equal to or greater than R
0	0.991025517	9.91025517E-01	1.000000000000
1	0.008934349	9.99959866E-01	0.008974482858

By using the same methodology and applying it to the SFPs for BFN Units 1 and 3 considering the total number of unverified plates in each SFP, the probability of a single missing plate in the SFPs can be calculated. As derived from Table 1, the BFN Unit 1 SFP contains 512 Boral plates that were not verified and BFN Unit 3 SFP contains 136 Boral plates that were not verified. The probability of a missing plate in BFN Unit 1 is 0.0287 providing 97.127% confidence that one or less plates are missing. The probability of a missing plate in BFN Unit 3 is 0.0078, providing 99.220% confidence that one or less plates are missing.

Note: The GEUMCO probability analysis utilized the number of verified plates as the sample size in the binomial distribution formula; however, the number of unverified plates should have been used. This resulted in incorrect, but conservative values. This conservatism resulted in a greater probability of having missing Boral plates, when in actuality the probability of missing plates is slightly less. This issue has been entered into the TVA Corrective Action Program (CAP).

c. Based on the results described above (response to item b) and the additional successful rack tests that have been performed at BFN alone, there is high confidence that all of the Boral plates were installed in the tubes and thus, in the HDSFRs.

As additional evidence that all of the Boral plates are installed, a Nuclear Regulatory Commission (NRC) special safety inspection in 1983 (documented in Inspection Report Nos. 50-259/83-18, 50-260/83-18 and 50-296/83-18, dated July 13, 1983) questioned whether all of the Boral plates had been installed in the HDSFRs at BFN. This inspection resulted in Notice of Violation (NOV 83-18). During an enforcement meeting held on June 1, 1983, the NRC documented the following.

> "It was brought out during the meeting that TVA has the Quality Assurance (QA) records that verify the correct Boral content of the fuel racks during manufacture and the licensee considers the Boral testing in the fuel pool backup certification of the fuel racks."

TVA provided these QA records to the NRC following the meeting and the NRC documented the following.

> "Subsequent to the meeting, the licensee provided copies of the QA records verifying correct Boral content. These records were reviewed by the senior resident and by Region II personnel and were found to be adequate indications for the safe use of the high density fuel racks."

These quotations indicate, at that time, TVA was in possession of documentation to satisfy the NRC reviewers that the BFN Unit 1 HDSFRs had sufficient Boral content and it was present in the racks.

Therefore, based on the evidence provided above, there is reasonable assurance that all of the Boral plates were properly installed in the BFN HDSFRs.

However, as discussed in the response to NRC RAI SFP-RAI 2 (submitted by TVA letter to NRC dated March 8, 2016), the failure to maintain complete permanent records of the Boral verification testing and the inconclusive testing of tubes in the storage modules has been entered into the TVA CAP. Actions will be taken to resolve these issues associated with the SFP storage module Boral plate configuration in accordance with the CAP. This condition is considered a nonconforming condition. As a result, the actions taken will be to restore compliance with the design and analysis basis. The actions from the associated CAP Condition Report (CR 1136812) require demonstrating Boral plates are installed in all required locations consistent with the CSA assumptions (i.e., no Boral plates are missing).

d. The BFN and Hatch HDSFRs were supplied by GEUMCO, a subsidiary of General Electric. All of the Boral plates and coupons were provided by a single supplier that had patented the process for manufacturing Boral and were the only supplier of record for that product during the time frame when the BFN and Hatch HDSFRs were manufactured. Based on GEUMCO Product Quality Certification records, all of the BFN racks were certified prior to the end of 1983.

A comparison of HDSFR documentation from BFN and the Hatch plant verified that the physical properties of the Hatch tubes are identical to the BFN tubes and that both construction and testing requirements reference the same GE specification (C5445-HDFSS). Because of the lattice design of the HDFSR racks, the Boral containing tubes are an integral part of the rack structure. The racks cannot be assembled without all of the tubes in place for a given rack dimension (i.e., 13x13 or 13x17). Each rack constructed under the C5445-HDFSS specification were built with the same connection details for rack assembly as were the Boral tubes. As such, assurance is provided that the requirements for fabrication of the Boral plates, the assembly of the racks and the testing performed for neutron absorption results were similar for both facilities. Therefore, the Hatch HDSFR testing results may be applied to the BFN SFP racks.

TVA does not have access to similar testing results from other sites. In response to this RAI, TVA contacted personnel at the Brunswick, Vermont Yankee, Hatch, Monticello, and Pilgrim plants in order to obtain information regarding initial HDSFR installation testing results. However, TVA was not successful in obtaining this information. The Hatch testing information that supported the analysis discussed in response to item b above was supplied from GEUMCO records.

Attachment 1

(SFP-RAI 11)

Markup Pages for Proposed Change to Technical Specification 4.3.1.1

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

~~b.~~ A nominal 6.563 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.2 of the FSAR;

b. $k_{eff} \leq 0.90$ if in a dry condition, or in the absence of moderator, as described in Section 10.2 of the FSAR; and

c. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and
c.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

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b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and
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(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

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c. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and
c.

(continued)

Attachment 2

(SFP-RAI 11)

Re-typed Pages for Proposed Change to Technical Specification 4.3.1.1

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and

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c. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and

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c. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3 of the FSAR; and

b. Fuel assemblies having a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration; and

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c. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

(continued)

Withhold from Public Disclosure Under 10 CFR 2.390

ENCLOSURE 2

ANP-3495P, Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2

(Proprietary version)

ENCLOSURE 3

ANP-3495NP, Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2

(Non-proprietary version)

AREVA

ANP-3495NP
Revision 1

Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2

Licensing Report

July 2016

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	Page 1-1	A footnote has been added.
2	Page 2-1	The response to SFP-3a has been updated to include TVA's response.

Contents

List of Tables

List of Figures

Nomenclature

Acronym	**Definition**
BFN	Browns Ferry Nuclear Plant
CLTP	Current Licensed Thermal Power (3458 MWt)
CSA	Criticality Safety Analysis
EPU	Extended Power Uprate
LAR	License Amendment Request
OLTP	Original Licensed Thermal Power (3293 MWt)
RAI	Request for Additional Information
REBOL	Reactivity Equivalent Beginning Of Life (lattice design)
SFSP	Spent Fuel Storage Pool
TS	Technical Specifications
TVA	Tennessee Valley Authority

1.0 Introduction

In Reference 1, the Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) to modify the operating license for the Browns Ferry Nuclear Plant (BFN) for an extended power uprate (EPU). The amendment, if approved, would allow for an increase in the licensed reactor thermal power from the current licensed thermal power (CLTP) of 3458 MWt to a new licensed thermal power of 3952 MWt, approximately 120% of the original licensed thermal power (OLTP) of 3293 MWt.

During a November 10, 2015 U. S. Nuclear Regulatory Commission (NRC) public meeting with TVA, the NRC requested a copy of the current spent fuel storage pool (SFSP) criticality safety analysis (CSA) report for BFN to support review of the EPU LAR. The requested report was provided as Enclosure 1 of Reference 2. The NRC staff has determined that additional information is needed to complete their review of the EPU LAR (Reference 3). This document addresses the subset of questions within Reference 3 identified as “RAIs Regarding SFP Criticality” and includes responses to the Request for Additional Information (RAI) that contains AREVA content[1].

References

1. Letter, JW Shea (TVA) to USNRC, “Proposed Technical Specifications Change to TS-505 – Request for License Amendments – Extended Power Uprate”, CNL-15-169, September 21, 2015. (Accession Number ML15282A152)

2. Letter, JW Shea (TVA) to USNRC, “Proposed Technical Specifications (TS) Change TS-505 – Request for License Amendments – Extended Power Uprate (EPU) – Supplement 1, Spent Fuel Pool Criticality Safety Analysis Information”, CNL-15-249, December 15, 2015. (Accession Number ML15351A097)

3. Letter, FE Saba (USNRC) to JW Shea (TVA), “Browns Ferry Nuclear Plant , Units 1, 2, amd 3 – Request For Additional Information Related To License Amendment Request Regarding Extended Power Uprate”, June 21, 2016. (Accession Number ML16154A544)

4. ANP-3160(P) Revision 1, *Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM Fuel*, AREVA Inc., December 2015. (provided as Enclosure 1 of Supplement 1 to the EPU LAR dated December 15, 2015)

[1] An exception to this is the response to SPF-RAI 3a, which is a response from TVA.

5. Rearden, B. T. and ORNL Staff, "Criticality Safety Enhancements for Scale 6.2 and Beyond", ICNC 2015, Charlotte, NC, September 2015, page 1255.

6. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

7. ANP-2860P Revision 2 Supplement 2P Revision 1, *Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate*, AREVA Inc., August 2015.

2.0 RAIs and Responses

2.1 *SFP-RAI 3*

The SFP criticality analysis documented in ANP-3160 used a 2x2 infinite array to determine the k-effective for normal conditions, then analyzed various accident conditions to determine the maximum increase in reactivity. This reactivity increase was then applied as an adder to the k-effective for normal conditions. Since the interface between two SFP rack modules includes face adjacent cells that are separated by stainless steel closure plates with no Boral, the 2x2 infinite array does not accurately capture the local configuration in these locations.

As part of the normal condition analysis, the reactivity for nominal rack module interface spacing was demonstrated to be less than the 2x2 infinite array. As part of the accident condition analysis, a configuration where the rack module interface spacing was significantly reduced was determined to result in an increase in reactivity. The staff notes that if there are no controls in place to ensure that the rack module spacing is equal to or greater than the nominal spacing, then a reduction in rack module spacing may become part of the normal condition. Since a sufficiently large reduction in the rack module spacing would result in a more reactive configuration, provide the following:

a. *A discussion of the controls in place at the plant to ensure that the nominal rack module spacing is maintained for normal operating conditions, or*

b. *If no such controls exist, a discussion of the reactivity impact of potential reductions in rack module spacing on the normal and accident conditions. For example, an accident configuration postulating a single missing Boral plate assumes that the missing plate is located on the interior of a SFP rack module. Other possible configurations involving missing Boral plates do not appear to have been analyzed. For example, a missing Boral plate might be located at the periphery of a SFP rack module, in such a location that results in three face adjacent fuel assemblies without Boral plates between them.*

Response:

a) The original design and installation documents for the high density fuel racks (HDSFR) provided nominal spacing requirements for the installation. Tennessee Valley Authority

(TVA) design controls provide assurance that installations comply with evaluated criteria. Whenever deviations from design requirements are discovered, Browns Ferry Nuclear Plant (BFN) resolves the issues through appropriate analyses and evaluations documented in the TVA corrective action program.

In the TVA submittal dated December 2, 1977 for license amendments 42, 39 and 16 for BFN Units 1, 2 and 3 (ML4008006709), the modules were analyzed for Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) conditions. BFN established that a limited amount of sliding may occur, and therefore rack spacing may be affected. In cases where sufficient spacing between racks exists, sliding up to 0.65 inches can be expected under OBE conditions. However, subsequent evaluations have shown that even when racks are in direct contact with adjacent racks it does not result in an unacceptable loss of criticality margin and K_{eff} remains within Technical Specification requirements.

BFN has seismic instrumentation in order to assess the effects of earthquakes on the plant which may occur that exceed the ground acceleration for the Operating Basis Earthquake (OBE = 0.10g ground acceleration). Annunciation is received in the control room when any one of the three accelerometers trigger indicating that it has sensed seismic motion in excess of 0.01g. Whenever OBE conditions are exceeded, all three BFN units are required to be shutdown for inspection and test activities. No documentation can be found indicating that a seismic motion alarm has been received (0.01g) at BFN that was associated with a validated seismic event. From this it can be concluded that no seismic activity has influenced the positions of any of the installed HDSFR modules since initial installation.

The TVA design and maintenance controls in place today through normal work control processes ensure that if rack configurations are changed by planned maintenance activities spacing will be maintained within design tolerances or resolved through engineering evaluation. Currently, no specific requirements exist to conduct inspections following OBE or SSE events. However, if racks are moved for any reason such that gaps are lost and direct rack-to-rack contact occurs the calculation discussed in SFP RAI-3b assures that K_{eff} remains within Technical Specification required values. Therefore, no additional controls are required.

b) In the introduction to the specific questions above, the statement is made that "…*the 2x2 infinite array does not accurately capture the local configuration in these locations".* While this statement is correct, it should also be noted that the use of the 2x2 rack model for the base reactivity used in the $k_{95/95}$ calculation is due to its demonstrated conservatism versus the more explicit individual rack models. Specifically, the 2x2 rack model provides greater than 14 mk conservatism to the explicit modeling of a 13x13 module as shown in Table 6.1 of ANP-3160P (0.8964 – 0.8819 = 0.0145). The same table also shows greater than 9 mk margin for the less explicit 13x13 model used in some of the abnormal/accident analyses. Additionally, the 13x13 rack model was shown to be similar to the co-resident 13x17 rack model in Table 6.3 of the same report. Consequently, the impact of less than nominal rack to rack spacing would need to exceed this level of conservatism before it would need to be explicitly accounted for in the $k_{95/95}$ calculation. Similarly, the $k_{95/95}$ calculation includes a 6 mk (0.006 Δk) adder to account for a missing Boral plate as the limiting accident in the Δk_{sys} term (Section 7.8 of Reference 4). A missing Boral plate on or near the edge of the rack would have to exceed this reactivity impact before it would need to be considered.

Additional calculations have been performed to specifically address the potential impact of a missing Boral plate on or near the edge of the 13x13 rack module. Two locations were chosen for a missing plate analysis; 1) a missing outside plate (i.e. between the racks), and 2) a missing adjacent plate (i.e. an inner plate in an edge cell). This second location is the same as that identified in the question (i.e. three face adjacent fuel assemblies). This second condition would be the expected limiting condition because it has the potential for higher reactivity due to closer spacing of the fuel assemblies with no neutron absorber between the assemblies. In contrast, the missing edge plate scenario still has absorber between the assemblies as part of the Boral tube in the adjacent rack.

The new analyses also include the impact of variable spacing between the racks. The variable spacing includes nominal rack spacing, the minimum spacing used in the original seismic accident analysis (less than ½ inch), and a new no spacing condition. The no spacing condition uses a minimal water gap that is consistent with the base plates of the racks being in contact with each other, while still modeling the nominal storage tube pitch. (This is conservative because the increased tube pitch needed to reduce the closure plate to closure plate gap from 0.461" to 0.120" is neglected). Adjustments were made to the original 13x13 boron-10 only model from ANP-3160P in order to provide a more accurate representation of the closure plates. Comparison for nominal spacing shows that the model provides similar results to that of the original model documented in Table 6.1 of ANP-3160P, i.e. k-effective of 0.8860 versus the original 0.8870.

The impact of the missing plate on or near the edge of the rack is summarized in Table 2. Results are provided for base analyses (i.e. no missing Boral plates) for each of the analyzed rack spacing conditions. The first conclusion from these base comparisons is that the limiting reactivity condition is the no rack spacing scenario. It is noted that even for this limiting spacing condition the k-effective value for the base condition of 0.8959 remains below the value of 0.897 used in the $k_{95/95}$ calculation. Therefore, no adjustments to the $k_{95/95}$ result would be required to account for less than nominal rack to rack spacing.

Table 2 also shows that the limiting missing Boral plate result occurs for this same spacing condition in which the base plates of the racks are in contact. The missing plate analyses in Table 1 assume the assemblies are centered within their individual cells. The limiting no rack spacing scenario was further evaluated in an attempt to optimize the reactivity change with alternate assembly placements within the cells adjacent to the missing plate. Specifically, a series of three additional cases were examined as summarized in Table 2. These analyses indicate a maximum impact of less than 2 mk (0.0015 Δk) for a missing Boral plate near the edge of the rack. The configuration for this limiting condition is illustrated in Figure 1. The reactivity impact for this configuration is significantly below the 6 mk (0.006 Δk) determined for an internal missing Boral plate and included as the limiting accident condition in the $k_{95/95}$ result through the Δk_{sys} term (Section 7.8 of Reference 4).

In summary, the original $k_{95/95}$ result of ANP-3160P requires no adjustment and remains applicable regardless of changes in either the rack to rack spacing or for a potential missing Boral plate located on or near the edge of the rack.

Table 1: Impact of Missing Boral Plate on Edge of Rack – Variable Rack Spacing

Description	k	Δk
Nominal Rack Spacing		
13x13 rack – no missing Boral plate	0.8860	--
13x13 rack – missing outside Boral plate	0.8867	0.0007
13x13 rack – missing adjacent Boral plate	0.8860	0.0000
Original Minimum Rack Spacing		
13x13 rack – no missing Boral plate	0.8915	--
13x13 rack – missing outside Boral plate	0.8917	0.0002
13x13 rack – missing adjacent Boral plate	0.8925	0.0010
No Rack Spacing [2]		
13x13 rack – no missing Boral plate	0.8959	--
13x13 rack – missing outside Boral plate	0.8959	0.0000
13x13 rack – missing adjacent Boral plate	0.8968	0.0009

Table 2: Optimized Reactivity Impact of Missing Boral Plate – No Rack Spacing

Description	**k**	**Δk**
Base Case – no missing Boral plate	0.8959	--
Missing adjacent plate – centered	0.8968	0.0009
Missing adjacent plate - assembly in non-Boral tube moved to missing plate	0.8974	0.0015
Missing adjacent plate - assembly in both non-Boral and Boral locations moved to missing plate	0.8956	-0.0003
Missing adjacent plate - assembly in non-Boral tube moved to missing plate and assembly in Boral tube moved ½ the distance to missing plate	0.8973	0.0014

2 No rack spacing indicates that the base plates of adjacent racks are in contact.

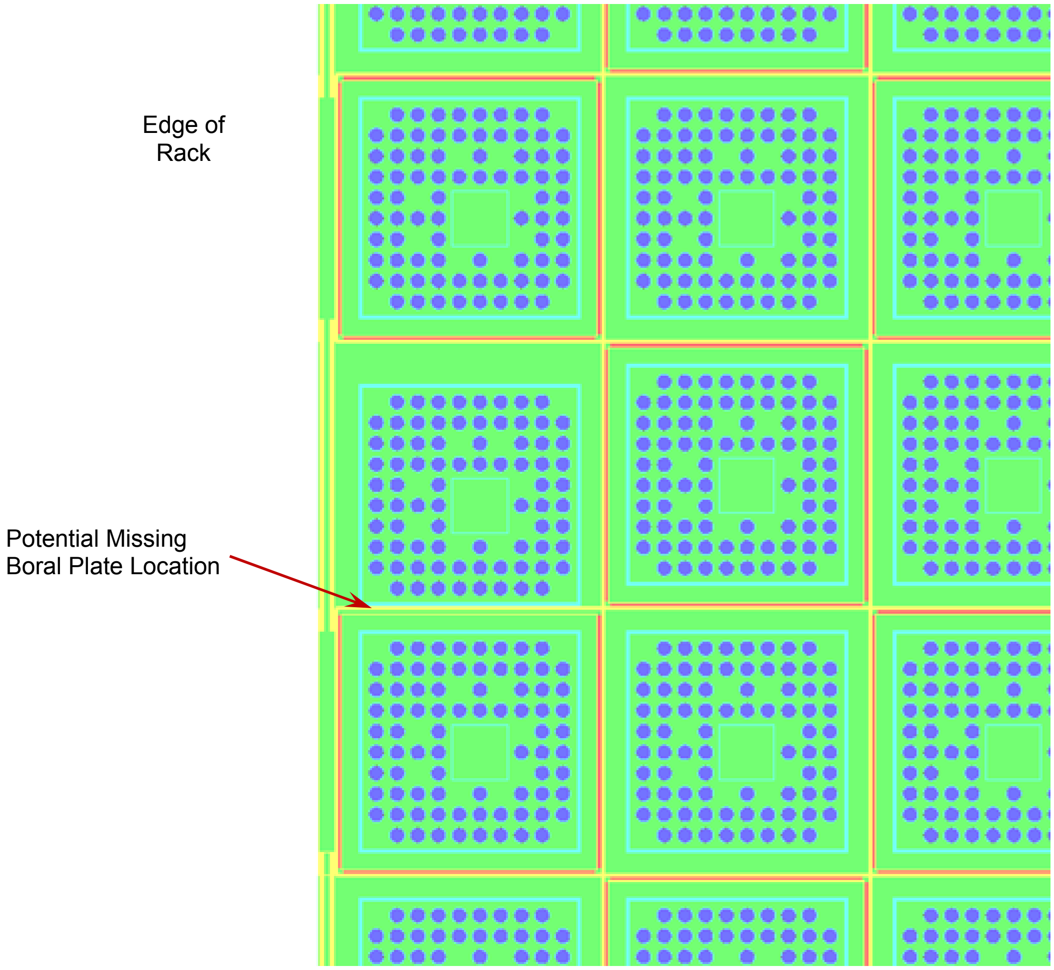


Figure 1: Potential Missing Boral Plate Near Edge of Rack (illustration only)

2.2 *SFP-RAI 4*

A number of calculations, including the final criticality analysis used to demonstrate compliance with the 10 CFR 50.68 requirement, are performed using the KENO V.a module in the SCALE 4.4a code system. ANP-3160 discusses how the hardware and software configuration is controlled to ensure that the code system used to run the calculations is appropriately qualified. However, no information is provided to describe how the licensee verified that each calculation converged to an appropriate solution. Appropriate code convergence must be verified to have reasonable assurance in the calculated k-effective results. Describe and justify the method used to verify appropriate convergence. Include the following factors: initial source distribution, *calculational parameters controlling the use of histories in computing the final calculated k-effective, convergence checks performed, and their acceptance criteria.*

Response:

Neutron source convergence was verified by examining the plotted eigenvalue histories to confirm that the average k-effective result is stable. In addition, the number of neutron generations skipped by KENO V.a was monitored. The actual fissile material is the fresh fuel assembly because the CSA utilizes reactivity equivalent beginning of life (REBOL) lattices. The assemblies are closely packed therefore a uniform source distribution is appropriate. The base calculation is actually a 2x2 array of bundles with periodic boundary conditions in the axial and radial directions.

As noted in Reference 5:

> *For many criticality safety applications, the additional step of performing a deterministic calculation to initialize the starting fission source distribution is not necessary. However, for challenging criticality safety analyses, such as as-loaded spent nuclear fuel transportation packages with a mixed loading of low and high-burnup fuel, even a low-fidelity deterministic solution for the fission source produces more reliable results than the typical starting distributions of uniform or cosine functions over the fissionable regions*

This criticality analysis would not be considered a challenging criticality safety analyses in that an infinite uniform geometry is used with a uniform burnup and enrichment.

The number of generations and neutrons per generation are evaluated to ensure that the eigenvalue has converged. These values are selected to ensure a converged eigenvalue rather than speed up the calculation. The calculations were performed with 2250 neutrons per generation and 450 tallied generations. Confirmation of convergence was repeated with the use of 3500 neutrons per generation with 700 tallied generations for the more limiting cases. Based on the near uniform fissile material, the array simplicity, and the confirmation evaluations, examination of the eigenvalue trend is adequate to assure proper convergence for this evaluation.

2.3 *SFP-RAI 5*

ANP-3160 states that the Boral is modeled using the design minimum B-10 areal density. The staff was unable to locate information describing how the Boral installed in the BFN SFP racks was verified to meet this minimum value, including any applicable measurement uncertainties. Provide an explanation of how the minimum B-10 areal density used in the SFP criticality analysis bounds all of the Boral plates installed in the BFN SFP racks.

Response:

TVA will provide a response separate from this document.

2.4 *SFP-RAI 6*

The spacer and channel growth due to burnup was accounted for in ANP-3160 by statistically combining the reactivity worth of a conservative estimate of the fuel rod pitch change and channel growth with other uncertainties (root of sum of squares method). This treatment is appropriate for a randomly distributed factor that is normally distributed around nominal dimensions, but spacer and channel growth would be expected to be normally distributed around an off-nominal value. Provide a justification for the treatment of the spacer and channel growth solely as an uncertainty.

Response:

AREVA concurs that the current treatment is a simplification that partially relies upon other conservatisms in the calculational approach. To address this request, the growth components have been re-evaluated to demonstrate that the original $k_{95/95}$ result is not adversely affected by this simplification.

In this alternate calculation, the manufacturing uncertainties have been revised to reflect the impact of irradiation growth as a bias rather than a manufacturing uncertainty. This involves revising the manufacturing uncertainties and adding a system bias term. The manufacturing reactivity uncertainties shown in Table 7.3 of ANP-3160(P) have been recalculated without the spacer growth component in the fuel rod pitch and without the channel growth. The new values are provided in Table 3.

The channel growth parameter was a reflection of the fuel channel bulge and it was modeled conservatively as an increase in the thickness of the fuel channel. In reality channel bulge will affect the shape of the fuel channel but has little effect on the thickness of the channel; therefore, the fissile to moderator conditions are not changed significantly. It is also noted that this analysis supports storage of both channeled and unchanneled fuel and that the flat 100 mil channel used in the base model contains more material than the advanced 100 mil fuel channel that is actually being used. (Table 6.2 of ANP-3160(P) shows the reactivity worth of different fuel channel options. The advanced 100 mil fuel channel has less material in the center than the flat channel). Therefore, channel bulge will not have any effect beyond what has already been modeled, for this reason it will be neglected in this alternate representation.

Calculations were performed to determine the reactivity worth and standard deviation values associated with spacer growth for inclusion as a system bias. These results are shown below.

[ ]

These impacts were included in the response to RAI-7 to determine the combined impact on the $k_{95/95}$ calculation. As shown in the response to RAI-7, the original $k_{95/95}$ result of 0.928 remains unchanged with the combined changes.

Table 3: Revised Manufacturing Reactivity Uncertainties

(Based upon BOL conditions using KENO V.a except as noted.)

2.5 ***SFP-RAI 7***

ANP-3160 states that pellet deformation with respect to burnup can be ignored, because it does not change the material content of the fuel. Fuel rod geometry changes as a result of irradiation may result in a change in the cladding outer diameter, which would change the fuel-moderator ratio. The results from the manufacturing tolerance analyses demonstrate that a change in the cladding outer diameter can have a positive reactivity impact. Provide further discussion on how fuel rod geometry changes might be expected to affect the criticality analysis.

Response:

ATRIUM 10XM fuel rod measurements indicate that the fuel rod diameter will creep down (decrease) from **[ ]** at beginning of life to about **[ ]** when the reference bounding lattices achieve maximum reactivity. Assuming the cross-sectional area of the clad is conserved, this reduction in fuel rod diameter produces a **[ ]** Δk increase.

[ ]

The $k_{95/95}$ calculation from Section 7.8 of ANP-3160(P) will now be recalculated with the modified manufacturing tolerance uncertainty (Δk_{tol} = 0.0073) from RAI 6 and with the growth dependent system bias results determined in RAI 6 and RAI 7.

$$k_{95/95} = k_{eff} + bias_m + \Delta k_{sys} + (C^2\sigma_k^2 + C_m^2\sigma_m^2 + C^2\sigma_{sys}^2 + \Delta k_{tol}^2)^{½},$$

where:

k_{eff}	=	Base in-rack reactivity from KENO V.a, (0.897)
$bias_m$	=	KENO V.a validation methodology bias (0.0075)
Δk_{sys}	=	Summation of applicable system variables (See the following table)
C	=	95% confidence level consistent with KENO V.a (2.0)
C_m	=	95/95 one-sided tolerance multiplier for a sample size of 68 (1.996)
σ_k	=	k-eff standard deviation from KENO V.a, (0.001)
σ_m	=	KENO V.a methodology uncertainty (0.0027)
σ_{sys}	=	$(\sigma_{sys1}^2 + \sigma_{sys2}^2 \ldots + \sigma_{sys\_n}^2)^{½}$, for Δk_{sys} uncertainties. (See the following table)
Δk_{tol}	=	Statistical combination of manufacturing reactivity uncertainties (**[ ]**)

[ ]

The resulting $k_{95/95}$ was then recalculated with a resulting $k_{95/95}$ of 0.9279. This continues to support the 0.928 $k_{95/95}$ result from the original evaluation.

2.6 *SFP-RAI 8*

In ANP-3160, Table 2.1, the second option to satisfy the fuel design limitations for enriched lattices involves use of a CASMO-4 storage rack model provided in Appendix A to evaluate each enriched lattice of a fuel assembly. The criticality analyses were performed using commercial grade uranium (CGU) fuel to conservatively bound blended low enriched uranium (BLEU) fuel, and the CASMO-4 model in Appendix A reflects this assumption. However, the previously manufactured BLEU fuel was modeled explicitly as BLEU fuel in the Appendix B evaluation of legacy fuel. Consequently, it is not clear if modeling a fuel lattice as CGU with the same uranium-235 (U-235) enrichment is required as part of satisfying the Table 2.1 limitations, or if the Appendix A inputs may be modified to explicitly model the BLEU fuel. Provide clarification on the intended application of this criterion. If explicit modeling of BLEU is allowed in meeting this criterion, then discuss how BLEU specific uncertainties are addressed, such as manufacturing tolerances, depletion uncertainties, and code validation uncertainties.

Response:

The intention of the second option in Table 2.1 of ANP-3160P is to allow for a direct calculation of in-rack k-infinity at peak reactivity conditions for any enriched lattice within a fuel assembly that does not meet the specified lattice enrichment or gadolinia loading requirements. The introduction to Appendix A indicates that the intent is to adapt the CASMO4 sample inputs provided in Tables A.1 and A.2 to model the fuel that does not conform to the enrichment and gadolinia requirements. This means that the specific lattice uranium enrichment and gadolinia loading inputs would be substituted for the reference bounding lattice inputs provided in the examples. Although not explicitly discussed, the intent was that Blended Low Enriched Uranium (BLEU) lattices could be explicitly modeled in the unlikely event that a direct reactivity calculation is required[5].

The impact of BLEU is addressed in Section 6.4 of ANP-3160P. The primary difference between BLEU and CGU fuel is the initial ^{234}U and ^{236}U content, both of which are explicitly included in the CASMO4 inputs for the BLEU lattices. Of these the ^{236}U isotope has the most significant impact on lattice reactivity because it acts as a neutron poison. Consequently, BLEU lattices exhibit less reactivity than a corresponding CGU lattice with the same enrichment. The reactivity impact of ^{236}U is illustrated in Figure 6.4 of ANP-3160P.

Manufacturing Uncertainty

As part of the BLEU evaluation provided in Section 6.4 of ANP-3160P, the statement was made that BLEU specific manufacturing uncertainties were not addressed due to the conservatism inherent in the use of the CGU lattice reactivity. However, the following provides a more direct approach showing that the CGU manufacturing uncertainty bounds a more explicit treatment of BLEU fuel.

[ ]

5 The maximum lattice average enrichment of 4.70 wt% ^{235}U is unlikely to be exceeded for a D-lattice ATRIUM 10XM design. It is considered to be similarly unlikely that a lattice would not meet the minimum gadolinia requirements, however; the ability to perform a direct calculation is included for these potential special cases.

Code Validation and Depletion Uncertainties

The BFN SFSP CSA is primarily a KENO based calculation analysis using CGU REBOL lattices. Consequently, there is no direct impact of BLEU modeling on the KENO validation bias or uncertainty. However, CASMO4 is used in the determination of the reference bounding lattices used in defining the REBOL lattices. CASMO4 may also potentially be used in the validation of actual lattice reactivity as discussed in Table 2.1 and Appendix A of ANP-3160P. The following discussion focuses on the potential impact of BLEU fuel on the CASMO4 validation and depletion uncertainty.

As summarized in Section D.4 of ANP-3160P, the validation (i.e., calculation) and depletion uncertainties were combined on a 95/95 basis resulting in a limiting combined uncertainty of **[ ]** Δk. This combined value was conservatively rounded up to 0.010 Δk and subsequently used as an adder in the definition of the REBOL lattices. The use of the calculation and depletion uncertainty as an adder (i.e. equivalent to a bias term) introduced an additional level of conservatism in the final $k_{95/95}$ result. Specifically, if this had been included as an uncertainty term instead of a bias the $k_{95/95}$ value would have decreased from 0.928 to 0.922, as described in the footnote on page 2-3 of ANP-3160P. Even if it were determined that BLEU fuel warranted a higher depletion uncertainty, the conservative treatment described above would remain bounding. However, as indicated in the following paragraph, historical data at BFN provides assurance that BLEU fuel does not increase the required depletion uncertainty.

The aforementioned combined CASMO4 validation and depletion uncertainty included an EMF-2158(P)(A) (Reference 6) depletion uncertainty component which is described in Section D.3.1 of ANP-3160P. It is recognized that higher than predicted reactivity values could result if CASMO4 were to mispredict the depletion of the ^{234}U and/or ^{236}U. Review of 12 cycles of Browns Ferry core follow with BLEU fuel show very consistent cold critical results. The continued applicability of AREVA methods to BFN at EPU conditions is supported in ANP-2860P Revision 2 Supplement 2P (Reference 7, included as Attachment 34 of the BFN EPU LAR). Figures 7-1 and 7-2 of Reference 7 provide plots of cold critical results that are primarily BLEU cycles. As noted on page D-19 of ANP-3160(P), the in-sequence cold critical results provide a good indicator of how depletion uncertainty could affect spent fuel pool criticality. These results indicate that the 0.003 Δk depletion uncertainty used in Section D.4 of ANP-3160(P) can also represent the BLEU fuel. Therefore, the combined validation and depletion uncertainty for BLEU fuel will not differ significantly from the results shown in Appendix D of ANP-3160(P).

In summary, the discussion above supports that the use of CASMO4 for Appendix A reactivity validation remains acceptable for BLEU fuel. This is based in part upon the continued applicability of the manufacturing uncertainty. It is also supported by BFN core follow history and the conservative approach in applying the CASMO4 validation and depletion uncertainties

2.7 *SFP-RAI 9*

ANP-3160 states that the gadolinia manufacturing uncertainty is evaluated with a combination of KENO V.a and CASMO-4. Since the final criticality analysis utilizes a REBOL lattice with no gadolinia, it is not clear how the gadolinia manufacturing uncertainty was determined using KENO V.a. CASMO-4 appears to be used primarily to compute a depletion based adder. Provide a description of the model used in KENO V.a to evaluate the gadolinia manufacturing uncertainty.

Response:

As shown in Figure 2.2 of ANP-3160(P) (Reference 4), the reference bounding assembly is comprised of lattices XMLCT-470UL-8G35 and XMLCB-470UL-8G3919. The gadolinia tolerance was calculated for 3.5% and 3.919% gadolinia. The larger tolerance value was then rounded up and subtracted from the associated gadolinia concentration, 3.919%. The 2x2 storage cell KENO model was then modified to include 8 gadolinia rods in the locations shown in Figure 2 (red representing the gadolinia rods). Calculations were performed with all gadolinia rods at 3.919% and then again using the lower concentration allowed by the tolerance. As indicated in Table 7.3 of ANP-3160(P), the beginning of life change in k-effective is very small. (The significant effect of the gadolinia concentration tolerance will occur at peak reactivity. This was represented with the CASMO-4 based depletion adder).

The 2x2 storage cell KENO model with 8 gadolinia rods (described above) was modified to represent the pellet density of the gadolinia rods at the minimum tolerance level. This condition increased the beginning of life k_∞ by 0.0012 Δk. CASMO-4 cases were also created to investigate the effect of the gadolinia pellet density tolerance at peak reactivity. Based on these results a small depletion based adder was also included.

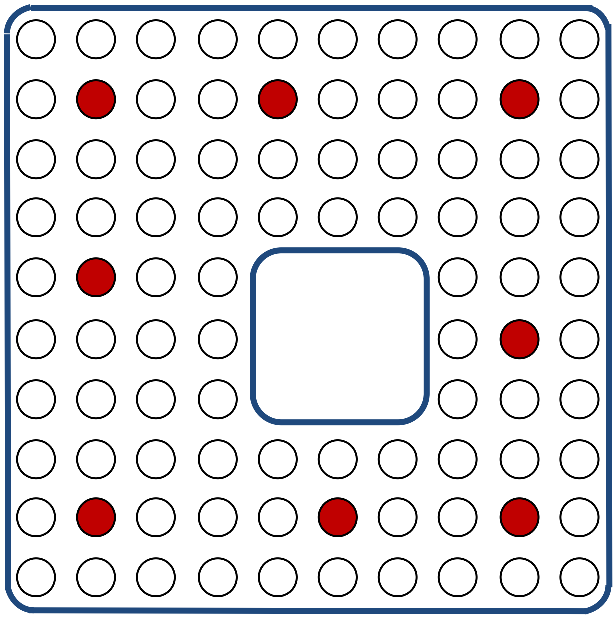


Figure 2: Gadolinia Pin Locations (based on the Bottom Reference Bounding Lattice)

2.8 *SFP-RAI 10*

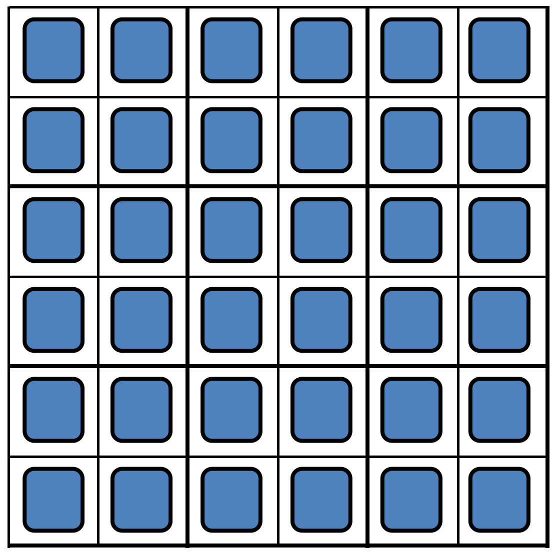
ANP-3160 discusses a series of calculations in which the radial location of the fuel lattice within the SFP cell was varied to account for fuel assembly lean. Provide additional detail on what configurations were analyzed.

Response:

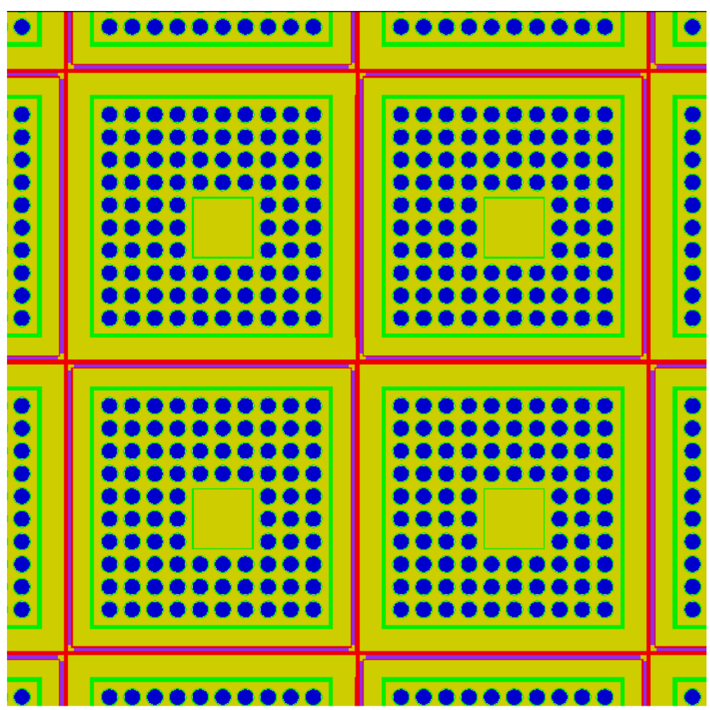
A 6x6 array of fuel assemblies (with periodic axial and radial boundary conditions) was selected to evaluate assembly lean. A base case and 14 other perturbation cases were created as described in the following table. Note the calculations performed model a translation of the assemblies instead of actual leaning assemblies. It is expected that this model will bound the response of a leaning assembly where only the upper section of the assembly will experience the change.

Table 4: Summary of Configurations Analyzed for Assembly Lean

Case	Illustration	Description
Base	Figure 3	6x6 assembly array – all assemblies centered within cell. The 6x6 array can also be visualized as a 3x3 array of groups of 4 assemblies each. i.e. 9 total groups of 4 assemblies.
Diagonally Inward	Figure 4	1, 2, 3, and then 4 of the center 4 assemblies were shifted diagonally inward while the surrounding 35 to 32 assemblies remain centered.
Half Diagonally Inward	Figure 5	The 4 center assemblies were shifted inward half the diagonal distance while the surrounding 32 assemblies remained centered.
Diagonally Outward	Figure 6	1, 2, 3, and then 4 of the center 4 assemblies were shifted diagonally outward while the surrounding 35 to 32 assemblies remain centered.
Half Diagonally Outward	Figure 7	4 center assemblies were shifted outward half the diagonal distance while the surrounding 32 assemblies remained centered.
Top and Bottom	Figure 8	Each top and bottom pair of assemblies are shifted to be adjacent to each other in the y direction while maintaining their standard spacing in the x direction.
Left and right	Figure 9	Each left and right pair of assemblies are shifted to be adjacent to each other in the x direction while maintaining their standard spacing in the y direction.
5 of 9	Figure 10	5 of the 9 groups of 4 assemblies are shifted diagonally outward the others remain centered. In terms of the 3x3 (i.e. 9) groups of four assemblies the corner and center groups are shifted.
All Clustered	Figure 11	All assemblies are shifted diagonally such that 9 groups of 4 assemblies are all face adjacent.

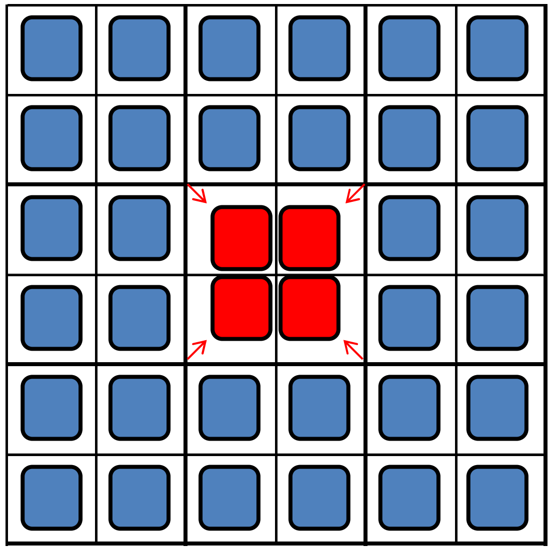


a) 6x6 Array of Assemblies

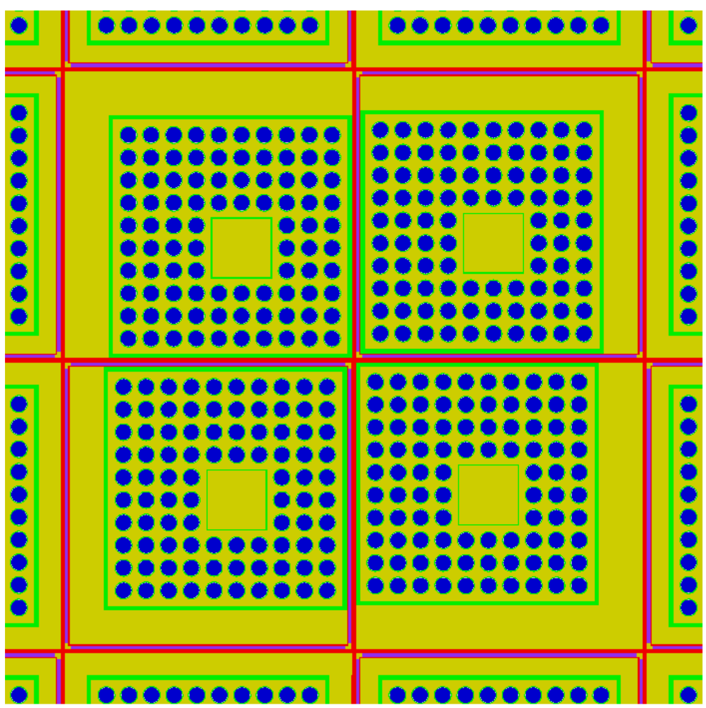


b) Center Group of 4 assemblies

Figure 3: Base Condition – Centered within Each Cell

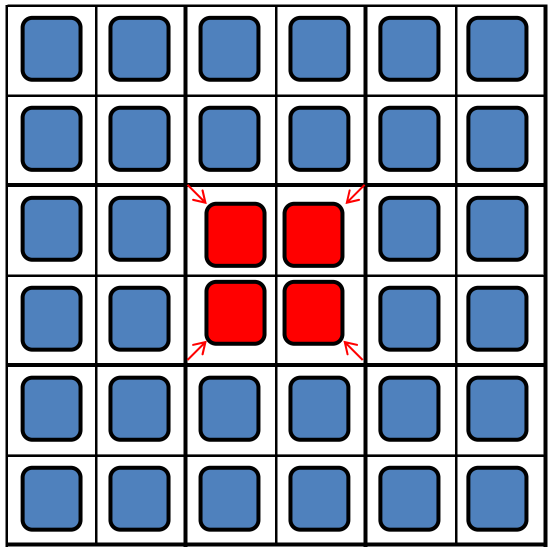


a) 6x6 Array of Assemblies

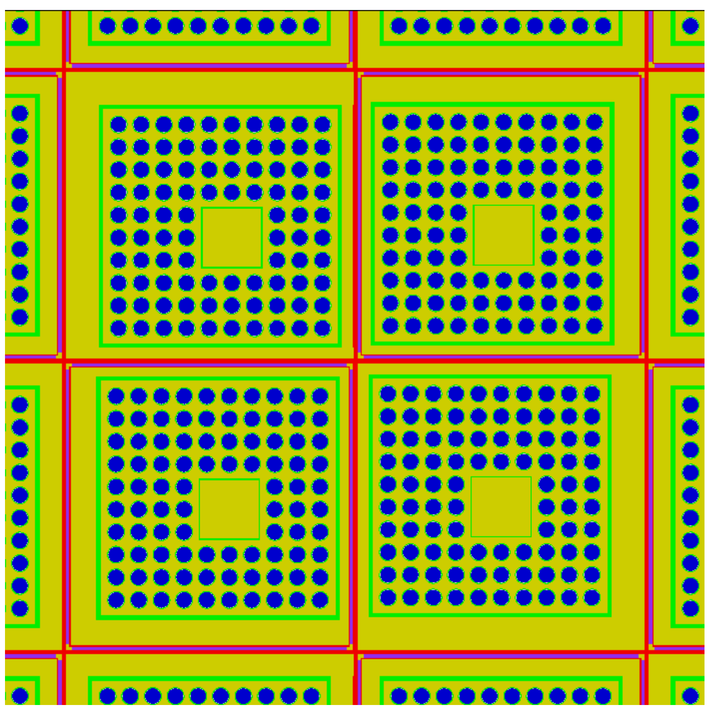


b) Center Group of 4 assemblies

Figure 4: Center Assemblies Shifted Diagonally Inward – Full Distance

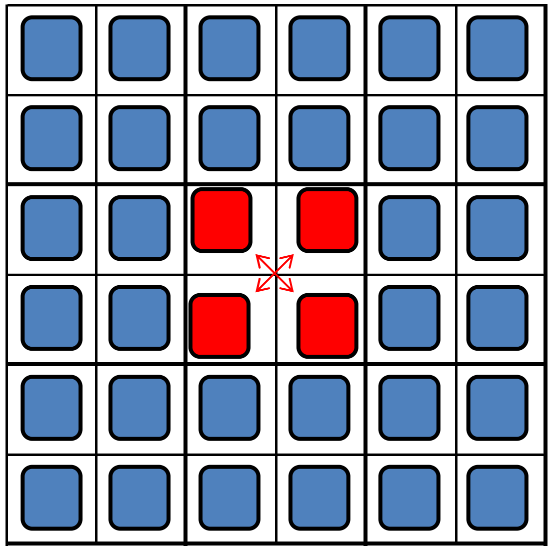


a) 6x6 Array of Assemblies

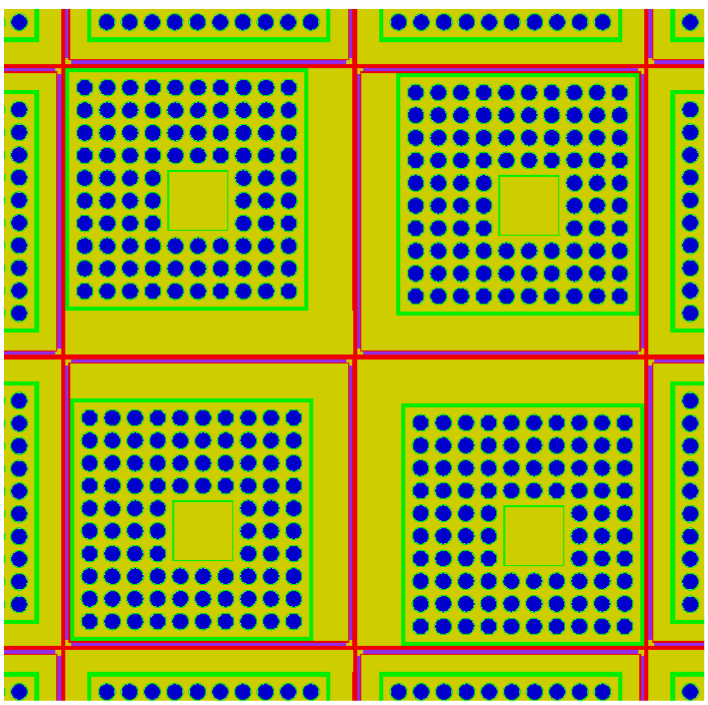


b) Center Group of 4 assemblies

Figure 5: Center Assemblies Shifted Diagonally Inward – Half Distance

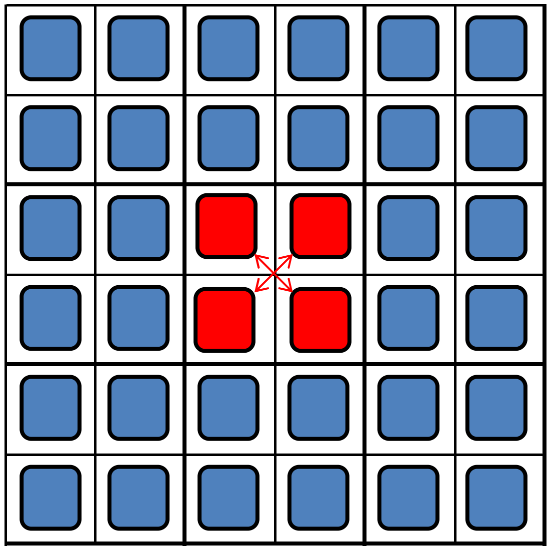


a) 6x6 Array of Assemblies

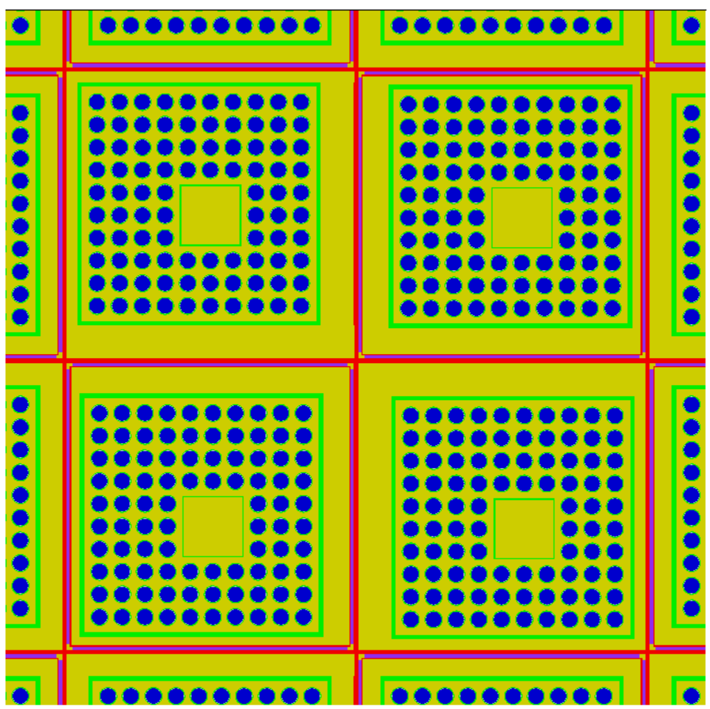


b) Center Group of 4 assemblies

Figure 6: Center Assemblies Shifted Diagonally Outward – Full Distance

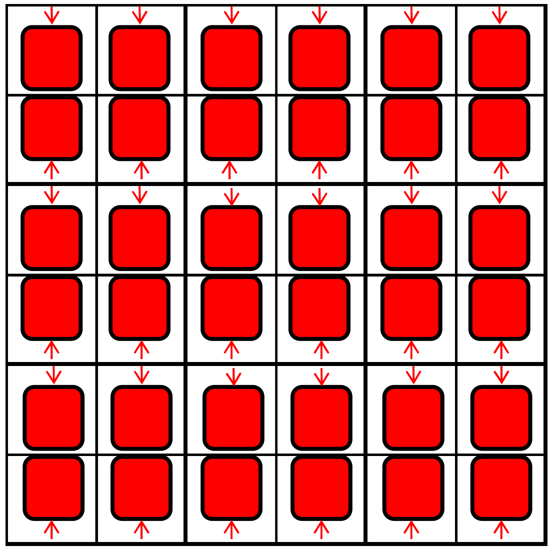


a) 6x6 Array of Assemblies

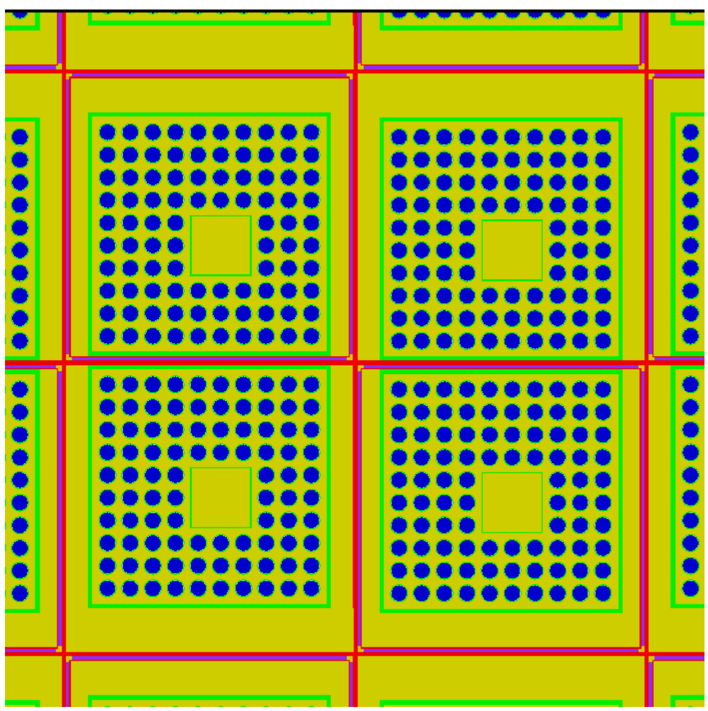


b) Center Group of 4 assemblies

Figure 7: Center Assemblies Shifted Diagonally Outward – Half Distance

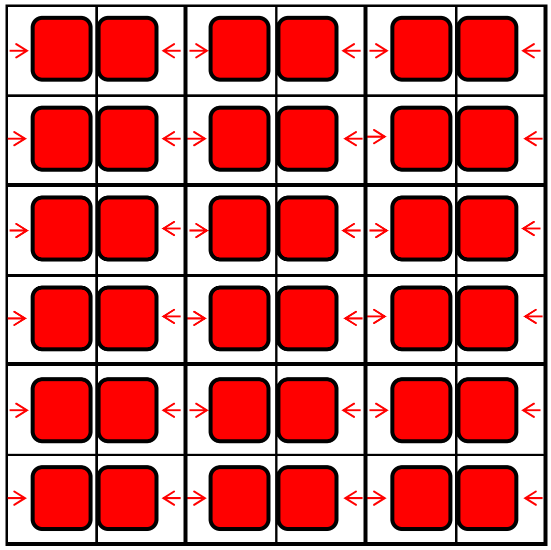


a) 6x6 Array of Assemblies

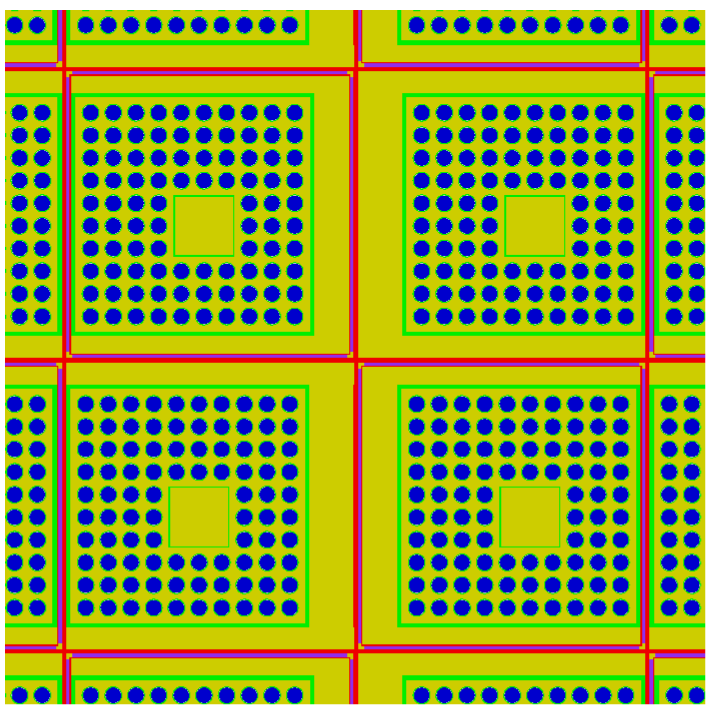


b) Center Group of 4 assemblies

Figure 8: Top and Bottom Center Assemblies Shifted Inward – Y Direction

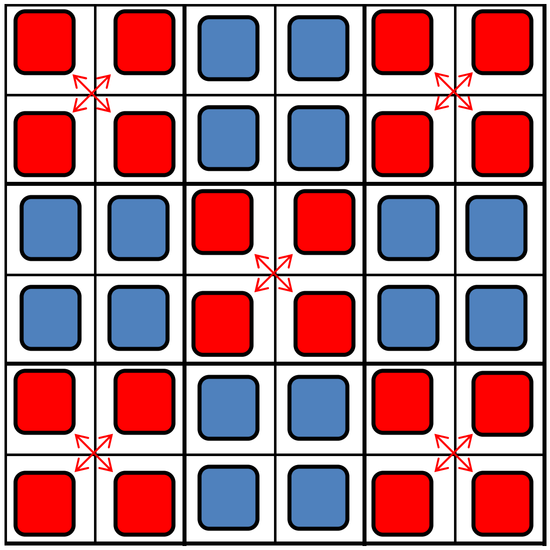


a) 6x6 Array of Assemblies

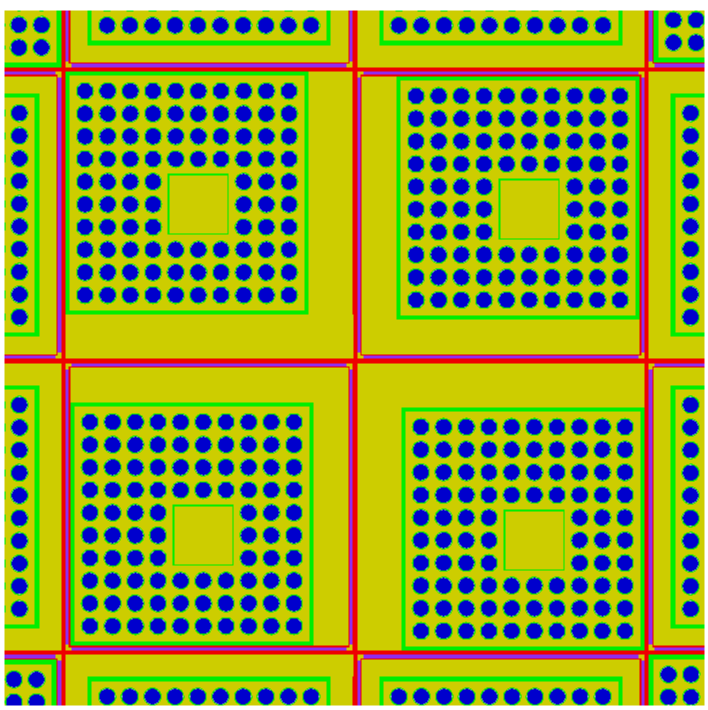


b) Center Group of 4 assemblies

Figure 9: Left and Right Center Assemblies Shifted Outward – X Direction

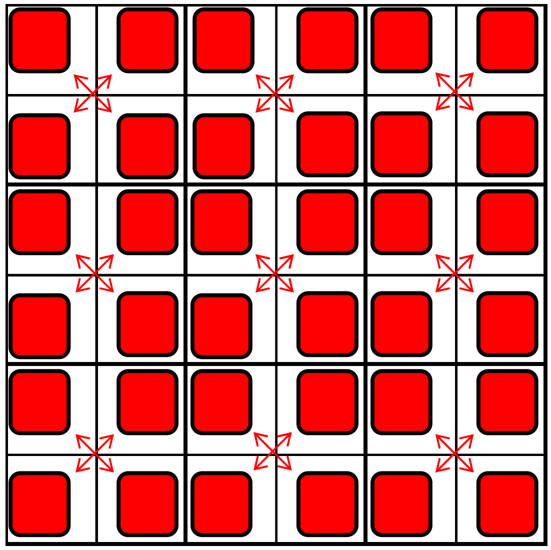


a) 6x6 Array of Assemblies

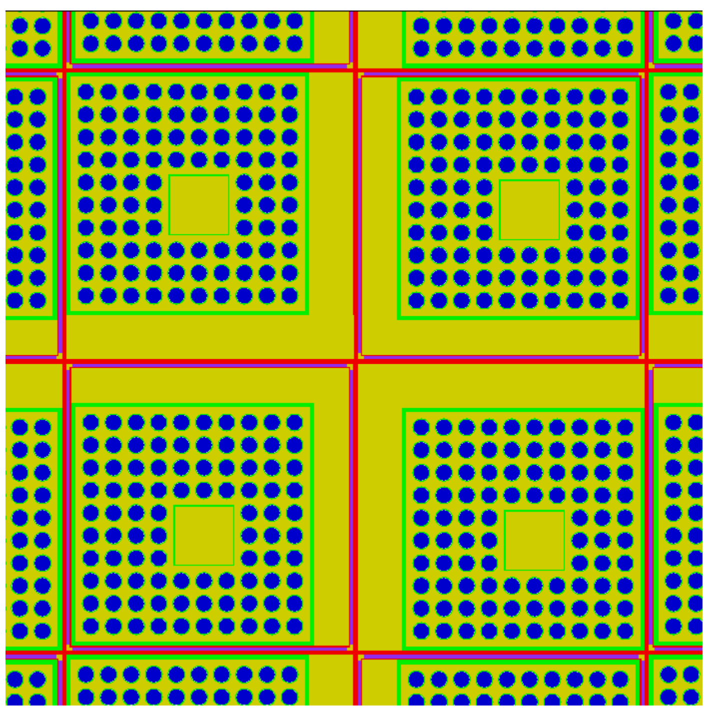


b) Center Group of 4 assemblies

Figure 10: Five of Nine Groups of 4 Assemblies Shifted Diagonally Outward



a) 6x6 Array of Assemblies



b) Center Group of 4 assemblies

Figure 11: All Assemblies Clustered in Groups of Four

2.9 *SFP-RAI 11*

The response to SFP-RAI 1 (ADAMS Accession No. ML16049A248) discusses requirements given in the criticality safety analysis (CSA) report (AREVA Report ANP-3160, Revision 1, attachment to ADAMS Accession No. ML15351A097) that must be met for fuel stored in the BFN SFPs. Fuel stored in the BFN SFPs may meet either one of two sets of requirements. The first set of requirements impose limits on the U-235 and gadolinia loading for the top and bottom lattices. The second set of requirements represent an in-rack k-infinity limit for the top and bottom lattices. The first set implicitly confirms that the second set is met, since the criticality analysis demonstrates that the maximum U-235 loadings combined with the minimum gadolinia loadings will result in an in-rack k-infinity that meets the provided limits. The NRC staff has determined that some of the margin to the regulatory limit as calculated in the CSA report will need to be credited to address potential nonconservatisms. Therefore, for the NRC staff to make a safety finding, the upper limit of the reactivity of the fuel lattices used in the CSA must be maintained. The Standard Technical Specifications list a limit on the k-infinity as calculated in the standard cold core geometry for fuel stored in SFPs, which is an example of one way of meeting this condition.

Propose a means of ensuring that the maximum reactivity of fuel assemblies loaded in the SFP will be limited such that the NRC safety finding for this CSA will continue to be valid for future fuel assemblies stored in the BFN SFPs.

Response:

TVA will provide a response separate from this document.

2.10 *SFP-RAI 12*

In response to SFP-RAI 2, the licensee indicated a review of plant records showed that the Boral verification testing was inconclusive for some cell locations on six BFN SFP storage modules. In order to evaluate these cell locations, a statistical analysis was performed to determine the probability that no more than one Boral plate is missing. The staff notes that if a single Boral plate is missing, then this would become part of the normal condition and should be evaluated as such. Since such a normal condition could result in a more limiting accident condition, provide the following:

a. The locations of the cells for which the Boral verification testing was inconclusive.

b. The statistical analysis performed to evaluate the cell locations for which the Boral verification testing was inconclusive.

c. A technical justification for the assumption that the normal BFN SFP rack condition does not include any missing Boral plates.

d. As part of an audit performed May 10 - 11, 2016, TVA provided draft documentation that included information to address one missing Boral panel as part of the normal condition. In addition to submitting this information on the docket, the NRC staff requests the following information:

i. A discussion of the applicability of the Edwin I. Hatch Nuclear Power Plant (Hatch) testing results to the BFN SFPs, more specifically addressing how the licensee has determined that the manufacturing process used to fabricate the SFP racks delivered to Hatch is the same as those delivered to BFN, and

ii. Information about any similar testing results from other sites that the licensee may be aware of.

Response:

a) TVA will provide a response separate from this document.

b) TVA will provide a response separate from this document.

c) TVA will provide a response separate from this document.

d) TVA will provide a response separate from this document.

ENCLOSURE 4

AREVA Affidavit

AFFIDAVIT

STATE OF NORTH CAROLINA)
) ss.
COUNTY OF MECKLENBURG)

1. My name is Thomas E Ryan. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA information contained in the AREVA document ANP-3495P, Revision 1, "Response to RAI for Browns Ferry Nuclear Plant EPU Submittal – SFSP Criticality Safety Analysis, Round 2," and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. 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 AREVA 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 AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA'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 AREVA.

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

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

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

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

8. AREVA 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.

Thomas E. Ryan

SUBSCRIBED before me this 12th

day of July, 2016.

Thomas A. Casias

Thomas A. Casias
NOTARY PUBLIC, STATE OF NORTH CAROLINA, COUNTY OF MECKLENBURG
MY COMMISSION EXPIRES: 15 Dec 2019
#200935100003

THOMAS A CASIAS
Notary Public, North Carolina
Mecklenburg County
My Commission Expires
December 15, 2019