Southern Nuclear

Regulatory Affairs

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WITHHOLD FROM PUBLIC DISCLOSURE UNDER 10 CFR 2.390 (DECONTROLLED UPON REMOVAL OF ATTACHMENT 2 OF ENCLOSURE 1)

September 13, 2022

NL-22-0609

Docket Nos.: 50-424
50-425

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

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

By letter dated June 30, 2022, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to allow for the use of lead test assemblies (LTAs) to demonstrate operating characteristics for accident-tolerant fuel (ATF).

By letter dated August 1, 2022 (ML22192A104), the U.S. Nuclear Regulatory Commission (NRC) staff notified SNC that requests for additional information (RAIs) were necessary early in their review. The NRC staff requested that SNC supplement the LAR to within 30 working days to keep the technical review on schedule.

Enclosure 1 Attachment 1 provides a response to RAI #1, which includes a reference to Enclosure 2 containing marked-up technical specification (TS) pages and Enclosure 3 containing clean-typed TS pages.
Enclosure 1 Attachment 2 provides proprietary responses to RAIs #2-6.
Enclosure 1 Attachment 3 provides non-proprietary responses to RAIs #2-6.
Enclosure 1 Attachment 4 provides the Westinghouse Affidavit, CAW-22-044, for Withholding Proprietary Information from Public Disclosure. The affidavit sets forth the basis upon which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-22-044 and should be addressed to Camille T. Zozula, Manager, Regulatory Compliance & Corporate Licensing, Westinghouse

Electric Company, 1000 Westinghouse Drive, Suite 165, Cranberry Township, Pennsylvania 16066.

In addition to the RAI responses, SNC seeks to correct typographical errors in the June 30, 2022 letter (ML22181B156) as follows. Westinghouse uses the term Optimized ZIRLO, which is an unregistered trademark, so the correct symbol is Optimized ZIRLO™. Westinghouse also uses the term ZIRLO, which is a registered trademark, so the correct symbol is ZIRLO®. In the June 30, 2022 letter, SNC reversed the symbols in error.

The conclusions of the No Significant Hazards Consideration Determination Analysis and Environmental Consideration contained in the LAR have been reviewed and are unaffected by these RAI responses.

This letter contains no NRC commitments.

In accordance with 10 CFR 50.91, SNC is notifying the state of Georgia of this license amendment RAI response by transmitting a copy of this letter to the designated state official.

If you have any questions, please contact Ryan Joyce at 205.992.6468.

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

Executed on September 13, 2022.

C. A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company

CAG/efb/cbg

Enclosure 1: SNC Response to Requests for Additional Information
Attachment 1: Response to Question 1
Attachment 2: Proprietary Response to Questions 2 – 6
Attachment 3: Non-Proprietary Response to Questions 2 – 6
Attachment 4: Affidavit
Enclosure 2: Revised Marked-up TS Pages
Enclosure 3: Revised Clean TS Pages

cc: Regional Administrator, Region II
NRR Project Manager – Vogtle 1&2
Senior Resident Inspector – Vogtle 1&2
State of Georgia Environmental Protection Division
RType: CVC7000

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

ENCLOSURE 1

SNC Response to Requests for Additional Information
Attachment 1
Response to Question 1

BACKGROUND

On January 27, 2022, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22028A046), a pre-submittal public meeting was held between the staff of the U.S. Nuclear Regulatory Commission (NRC) and representatives of Southern Nuclear Operating Company, Inc. (SNC). SNC stated that the proposed license amendment request (LAR) will revise (1) Technical Specification (TS) 4.2.1, “Fuel Assemblies,” and (2) TS 4.3.1, “Criticality.” SNC stated the proposed LAR could potentially revise TS 3.7.18, “Fuel Assembly Storage in the Fuel Storage Pool.”

On May 11, 2022 (ML22132A010), a partial open and partial closed pre-submittal meeting was held between the NRC staff and representatives of SNC. SNC presented slides 8 and 9 (ML22126A001) regarding a proposed license condition. During that meeting, the NRC staff expressed concerns with the proposed license condition in the absence of conforming changes to the applicable TS, and the NRC staff informed the licensee that the license condition and TS could not be inconsistent and that conforming changes were likely needed, as described in the January pre-submittal meeting discussions.

REQUEST FOR ADDITIONAL INFORMATION

Introduction

By letter dated June 30, 2022 (ML22181B156), Southern Nuclear Operating Company (SNC, the licensee) submitted a LAR for Vogtle Electric Generating Plant (Vogtle), Units 1 and 2. The proposed LAR would allow the use of lead test assemblies (LTAs) for accident tolerant fuel (ATF).

During its acceptance review of the LAR, the NRC staff requests that SNC supplement the application to address the information requested below, within 30 working days from the date of this letter. Timely submittal of the requested information would help keep the technical review on schedule.

Question 1 Regulatory Basis

The regulation in Section 50.36(c)(4) of the Title 10 of the *Code of Federal Regulations* (10 CFR) describes what Design Features information shall be included in TSs. The regulation states: “Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.” The information included in the proposed license condition appears to meet the definition of design features.

Question 1

The Vogtle Renewed Facility Operating License (RFOL) states, in part, that:

> C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the

Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 214 [for Unit 1, and Amendment No. 197 for Unit 2], and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

TS 4.2.1, “Fuel Assemblies,” states:

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (U02) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

TS 4.3.1, “Criticality,” states:

(Unit 1)

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. Keff < 1.0 when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. Keff < 0.95 when fully flooded with water borated to 511 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the “acceptable burnup domain” of Figures 3.7.18-1 or satisfying a minimum Integral Fuel Burnable Absorber (IFBA) requirement as shown in Figure 4.3.1-7 may be allowed unrestricted storage in the Unit 1 fuel storage pool.

e. New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 1 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

Interfaces between storage configurations in the Unit 1 fuel storage pool shall be in compliance with Figure 4.3.1-3. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-1, or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-7. "B" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235.

f. A nominal 10.25 inch center to center pitch in the Unit 1 high density fuel storage racks.

(Unit 2)

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. Keff < 1.0 when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. Keff < 0.95 when fully flooded with water borated to 394 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2 may be allowed unrestricted storage in the Unit 2 fuel storage pool.

e. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8 may be stored in the Unit 2 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown In Figure 4.3.1-1.

New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 2 fuel storage pool in a 2-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10 may be stored in the Unit 2 fuel storage pool as "low enrichment' fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2. New or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments may be stored in the Unit 2 fuel storage pool as "high enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2.

Interfaces between storage configurations in the Unit 2 fuel storage pool shall be in compliance with Figures 4.3.1-3, 4.3.1-4, 4.3.1-5, and 4.3.1-6. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2. "B" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8. "C" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235. "L" assemblies are new or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10. “H” assemblies are new or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments.

f. A nominal 10.58-inch center to center pitch in the north-south direction and a nominal 10.4-inch center to center pitch in the east-west direction in the Unit 2 high density fuel storage racks.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. Keff < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. Keff < 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

TS 3.7.18, “Fuel Assembly Storage in the Fuel Storage Pool,” states, in part:

> LCO 3.7.18 The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

TS 5.6.5, “Core Operating Limits Report (COLR),” states, in part, that:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1 “SHUTDOWN MARGIN”
LCO 3.1.3 “Moderator Temperature Coefficient”
LCO 3.1.5 “Shutdown Bank Insertion Limits”
LCO 3.1.6 “Control Bank Insertion Limits”
LCO 3.2.1 “Heat Flux Hot Channel Factor”
LCO 3.2.2 “Nuclear Enthalpy Rise Hot Channel Factor”
LCO 3.2.3 “Axial Flux Difference”
LCO 3.9.1 “Boron Concentration”

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

The following provisions are, in part, the proposed license conditions in the LAR.

Lead test assemblies (LTAs) 7ST1, 7ST2, 7ST3, and 7ST4 contain fuel rods that include advanced coated cladding features and doped or standard fuel material. Each of the four LTAs may contain up to four fuel rods with a maximum nominal U-235 enrichment of 6.0 weight percent; the maximum nominal U-235 enrichment of th 2 0 0 weight percent.

In lieu of the requirements in Technical Specification (TS) Section 4.2, the LTAs are permitted to be placed in limiting core regions for up to two cycles of operation without completion of representative testing.

In lieu of the requirements in TS Section 4.3, the LTAs are subject to the following alternate requirements:

1. These LTAs may be stored in the spent fuel storage racks as specified below:

    a. TS 4.3.1.2.b and 4.3.1.2.c must be met

    b. Storage in the Unit 1 and Unit 2 spent fuel racks is prohibited except:

        i. Unrestricted storage is allowed in the Unit 2 2-out-of-4 checkerboard storage configuration as shown in TS Figure 4.3.1-1.

        ii. Storage is allowed in the Unit 2 all-cell storage configuration (“A” assemblies as shown on TS Figures 4.3.1-3 and 4.3.1-5) when the LTAs reach 64,000 MWd/MTU of burnup.

2. These LTAs may be stored in the new fuel storage racks.

    Limiting Condition for Operation (LCO) 3.7.18 shall be considered met for the LTAs provided the alternate Section 4.3 requirements are met.

As previously noted by the NRC staff during the pre-submittal meeting held on May 11, 2022, the proposed license condition for the LTAs, as written, could be read as contradictory to and incompatible with the RFOL license condition C.(2), and subsequently TSs 4.2.1, 4.3.1, 3.7.18, and 5.6.5. Since exceptions applicable to the LTAs are only described in the proposed license condition and not in the applicable TS sections, a reader could interpret the more limiting condition in the TSs to take precedence over the proposed license condition, or vice versa. Please provide any supplemental information to the application, as appropriate, to explain or resolve this contradiction.

Since the information in the proposed license condition appears to meet the definition of Design Features, please explain how the regulatory requirements of 10 CFR 50.36(c)(4) will be met if the information in the proposed license condition is not incorporated in the TSs.

SNC Response to Question 1:

Upon further consideration, SNC is revising the request to describe the lead test assembly (LTA) provision for LTAs 7ST1, 7ST2, 7ST3, and 7ST4 directly in Technical Specification (TS) 3.7.18, 4.2 and 4.3, rather than rely on an updated Appendix D of the facility operating license (FOL). SNC hereby withdraws the Appendix D FOL markups provided in the June 30, 2022 request. The updated TS markups are provided in Enclosure 2 and clean-typed TS pages are provided in Enclosure 3. As discussed in the SNC response to Question 2, these LTAs are not generating operating limits, and as such, do not warrant changes to TS 5.6.5.

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
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ENCLOSURE 1

SNC Response to Requests for Additional Information
Attachment 3
Non-Proprietary Response to Questions 2-6

RAI 2:

In its letter dated June 30, 2022 (ML22181B156), SNC indicated it is using Westinghouse Topical Reports (TRs) (1) WCAP-18546-P/NP, "Westinghouse AXIOM® Cladding for Use in Pressurized Water Reactor Fuel" (ML21090A110), and (2) WCAP-18482-P/WCAP-18482-NP, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel" (ML20132A014). Confirm whether these TRs are being used to generate operating limits, and if so, how they are addressed in administrative controls.

Response:

The Westinghouse Topical Reports (TRs) WCAP-18546-P and WCAP-18482-P are not used to generate operating limits for reload designs using the Lead Test Assemblies. Therefore, no administrative controls need to be implemented regarding the use of these TRs. Only the methods described in the documents listed in Technical Specification 5.6.5.b will be used to generate the core operating limits contained in the Core Operating Limits Report.

Reference:

1. "VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 - ACCEPTANCE OF REQUESTED LICENSING ACTION RE: AMENDMENT REQUEST; APPLICATION TO ALLOW USE OF LEAD TEST ASSEMBLIES FOR ACCIDENT TOLERANT FUEL WITH REQUEST FOR ADDITIONAL INFORMATION (EPID L-2022-LLA-0097)," NRC ADAMS Accession Number ML22192A104.

Question 3 Regulatory Basis

General Design Criteria 35 (GDC 35), "Emergency core cooling," states, in part, that:

> *A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*

The regulation 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems (ECCSs) for light-water nuclear power reactors," requires nuclear power reactors fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding to be provided with an ECCS with certain performance requirements and ensures compliance with GDC 35.

The analysis of loss-of-coolant accidents (LOCAs) described in the LAR ensures compliance with 10 CFR 50.46 and GDC 35. Thermal hydraulic (T/H) design methods are used to analyze the outcome of a postulated LOCA.

The LAR makes the following statement:

> *The T/H design methods for the chromium coated cladding evaluation were reviewed in accordance with the NRC interim guidance. No modification or update to any NRC-approved topical reports on DNB correlations and thermal-hydraulic analysis methods is needed for applications to the LTA coated fuel rods.*

The referred-to NRC chromium coated cladding interim staff guidance document, ATF-ISG-2020-01, "Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept, Interim Staff Guidance, January 2020 (ML19343A121), contains a discussion of cladding thermal emissivity, and states the following:

> *Some system codes and accident analysis codes account for cladding surface emissivity and radiation heat transfer from fuel rods to other reactor core components, as well as radiation heat transfer to steam. In general, shinier surfaces have lower emissivity and therefore lower radiative heat transfer. As chromium coatings resist oxidation and retain their surface appearance, it is likely that the coating will negatively impact cladding temperature for transients where radiation to steam is the dominant mode of heat transfer. Therefore, it is likely necessary to revise the outer surface emissivity for accident analyses. This would apply equally to metallic and ceramic coatings (Seshadri, Philips, & Shirvan, 2018, (Reference 8)) [Seshadri, A., Philips, B., & Shirvan, K. (2018). Towards Understanding the Effects of Irradiation on Quenching Heat Transfer. International Journal of Heat Transfer.].*

Question 3

In a previously approved LAR for LTAs with coated cladding, thermal emissivity of the cladding was considered during the LOCA analysis, as mentioned in that safety evaluation (ML20363A242).

Please describe how the impact of thermal emissivity of the coated cladding has been dispositioned in the LOCA analysis performed for the licensee's proposed LAR for LTAs.

Response to Question 3

The small break (SB) and large break (LB) loss-of-coolant accident (LOCA) evaluation models (EMs) licensed for Vogtle Electric Generating Plant (Vogtle) Units 1 and 2 per the requirements of Appendix K to Title 10 of the Code of Federal Regulations (CFR) Part 50 were reviewed with respect to the behavior of chromium coated cladding to support the license amendment request (LAR) for operation of four accident tolerant fuel (ATF) lead test assemblies (LTAs). Using thermal and mechanical properties described in open literature for chromium coating material and based on test data available from the ongoing Westinghouse chromium coated cladding development program, the review [

]a,c to estimate the effect of the operation of the Vogtle LTAs up to the licensed burnup limit. In addition to cladding emissivity described below, the review considered the behaviors and properties summarized here with respect to the NOTRUMP EM and the BASH EM:

- Clad swelling and rupture: Section 3.10.1 of the LAR indicates Westinghouse burst testing results [

  ]a,c to assess the chromium coated cladding of the LTA.
- High temperature oxidation: Section 3.10.1 of the LAR indicates Westinghouse high temperature steam oxidation testing of chromium coated Optimized ZIRLO cladding [
  ]a,c.
  Appendix K to 10 CFR 50 requires use of the Baker-Just equation to calculate cladding oxidation from the metal/water reaction, which is conservative for chromium coated cladding.
- Material properties: [
  ]a,c.
  - Based on the heat capacity of chromium provided in Reference 6, the specific heat of chromium is generally [

    ]a,c specific heat of the chromium layer in the LTAs.
  - The thermal conductivity of chromium based on Reference 7 is [

    ]a,c to assess the LTAs.
  - The room temperature density of chromium based on Reference 8 is [

    ]a,c to assess the LTAs.
  - Thermal and elastic expansion [

    ]a,c to assess the LTAs.
- High temperature creep: Westinghouse creep strain testing [

  ]a,c.

For cladding emissivity, the values modeled in the NOTRUMP EM and BASH EM are based on the model described in the LOCTA-IV code (Reference 4) and reflect the oxidized surface of the zirconium alloy cladding. Based on Reference 4, the cladding emissivity of oxidized zirconium for the NOTRUMP EM and BASH EM ranges from 0.6 (lower bound) to [ ]^{a,c}. However, since the chromium coating resists oxidation, as noted in the regulatory basis section accompanying this question, the emissivity of the metallic chromium surface should also be considered, especially early in reactor operation. Per Reference 13, the emissivity of polished chromium ranges from 0.08 to 0.40. Westinghouse high temperature oxidation testing discussed previously and information available in open literature (References 9 through 12) indicates the surface of chromium coated cladding with various application techniques (cold spray, physical vapor deposition (PVD), arc ion plating, and atmospheric plasma spraying) will be oxidized under LOCA conditions. Reference 5 lists the emissivity for chromium oxide as 0.69 and 0.91, [ ]^{a,c}.

The review concludes that [ ]^{a,c} in the LOCA evaluation of the LTAs described in Section 3.5 of the LAR.

References

1. Westinghouse Topical Report WCAP-10079-P-A, Revision 0, “NOTRUMP A Nodal Transient Small Break and General Network Code,” August 1985.
2. Westinghouse Topical Report WCAP-10054-P-A, Revision 0, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” August 1985.
3. Westinghouse Topical Report WCAP-10266-P-A, Revision 2, “The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code,” March 1987.
4. Westinghouse Topical Report WCAP-8301, Revision 0, “LOCTA-IV Program: Loss-of-Coolant Transient Analysis,” June 1974.
5. Greenhut, V. A., “Effects of Composition, Processing, and Structures on Properties of Ceramics and Glasses,” in *ASM Handbook Volume 20 Materials Selection and Design*, edited by Dieter, G. E. (ASM International, 1997), 428.
6. Chase, M. W., Jr., “NIST-JANAF Thermochemical Tables, Fourth Edition,” *Journal of Physical and Chemical Reference Data, Monograph 9* 1 (1988); Data compiled in NIST Chemistry WebBook, Standard Reference Database Number 69, “chromium.”
7. Moore, J. P. et al., “Thermal conductivity, electrical resistivity, and Seebeck coefficient of high purity chromium from 280 to 1000 K,” *Journal of Applied Physics* 48, no. 2 (1977): 610.
8. Holzwarth, U et al., “Mechanical and thermomechanical properties of commercially pure chromium and chromium alloys,” *Journal of Nuclear Materials* 300, no. 2-3 (2002): 161-177.
9. Park, J-H et al., “High temperature steam-oxidation behavior of arc ion plated Cr coatings for accident tolerant fuel claddings,” *Surface and Coatings Technology* 280 (October 2015): 256-259.
10. Wang, Y. et al., “Behavior of plasma sprayed Cr coatings and FeCrAl coatings on Zr fuel cladding under loss-of-coolant accident conditions,” *Surface and Coatings Technology* 344 (June 2018): 141-148.
11. Brachet, J-C et al., “Early studies on Cr-Coated Zircaloy-4 as enhanced accident tolerant nuclear fuel cladding for light water reactors,” *Journal of Nuclear Materials* 517 (April 2019): 268-285.
12. Yeom, H. et al., “High temperature oxidation and microstructural evolution of cold spray chromium coatings on Zircaloy-4 in steam environments,” *Journal of Nuclear Materials* 526 (December 2019): 151737.

13. Howell, J. R. et al., *Thermal Radiation Heat Transfer* (Florida: CRC Press, 2011).
14. Westinghouse Topical Report WCAP-12610-P-A, Revision 0, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995
15. Westinghouse Topical Report WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, Revision 0, "Optimized ZIRLO™," July 2006.
16. Stephens, J. R. et al., "High-temperature creep of polycrystalline chromium," *Journal of the Less Common Metals* 27, no. 1 (April 1972): 87-94

NRC RAI No. 4:

The licensee is justifying storage of the LTAs in its New Fuel Storage Racks (NFSR) by comparison to its Analysis of Record (AOR). However, SNC did not specify a reference for its NFSR AOR and contrary to Vogtle, Units 1 and 2, TS 4.3.1.3.b and TS 4.3.1.3.c, the NRC staff did not find a discussion of Vogtle's NFSR biases and uncertainties in Section 4.3 of the Vogtle FSAR. To allow the NRC staff to evaluate the licensee's application against the criteria of 10 CFR 50.68(b)(2) and 10CFR 50.68(b)(2), please provide the appropriate reference for Vogtle, Units 1 and 2, AOR for its NFSR.

Response:

The appropriate reference for Vogtle Units 1 and 2 NFSR AOR is docketed as Enclosure 2 (Reference 1) of Reference 2. The inclusion of the following additional information provides more detail about the LTA specific analysis, indicating that the NFSR analysis does not rely on the AOR from a technical standpoint. [ ][a,c] Table 1 provides a bias and uncertainty rackup for the NFSR modeling discussed in Reference 3 for the fully flooded case, the optimum moderation case with IFBA and the optimum moderation case without IFBA.

Table 1: Bias and Uncertainty Table for the NFSR Analysis [a,c]

[ ]

[

]a,c Table 1 indicates storage acceptability without IFBA, while significant additional reactivity hold down is observed with IFBA present.

References:

1. Attachment 2 of Enclosure 2 of ELV-00511 (ML20244D565), “Criticality Analysis of Vogtle Fresh Fuel Racks.”

2. ELV-00511 (ML20244D557) “Application for amends to Licenses NPF-68 and NPF-81, revising Tech Spec 5.3.1 to change max enrichment to 4.55 weight percent U-235.”

3. ML22192A104, “VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – ACCEPTANCE OF REQUESTED LICENSING ACTION RE: AMENDMENT REQUEST; APPLICATION TO ALLOW USE OF LEAD TEST ASSEMBLIES FOR ACCIDENT TOLERANT FUEL WITH REQUEST FOR ADDITIONAL INFORMATION (EPID L-2022-LLA-0097).”

<u>NRC RAI No. 5:</u>

SNC's evaluation of the acceptability to store the LTAs in its NFSR, in its Vogtle, Unit 2, SFP two-out-of-four (2oo4) storage configuration (a repeating 2x2 array of storage cells where fuel assemblies are checkerboarded with empty storage cells), and demonstrating an acceptable response to the identified limiting accident in the SFP are all primarily dependent on the integral fuel burnable absorber (IFBA) loading in the LTAs. However, the IFBA loading is not included in the proposed license condition description of the LTAs. The NRC staff's review will rely on the IFBA loading in the LTAs. Describe the controls SNC will use to ensure that the IFBA loading used in the submittal will be met or exceeded in all respects.

<u>Response:</u>

The number of IFBA rods and details regarding loading and IFBA length are administratively controlled by both Westinghouse and SNC in accordance with the reload procedures. As part of this process, both Westinghouse and SNC will document confirmation that the as-built LTA IFBA characteristics meet the associated criticality analysis criteria (e.g., Minimum 128 IFBA rods per LTA, Minimum 1.5X nominal loading and maximum nominal 8 inch cutback).

<u>NRC RAI No. 6:</u>

The licensee is proposing to allow storage of the depleted LTAs in its Vogtle, Unit 2, SFP All Cell storage configuration, a repeating 2x2 array of storage cells where all fuel assemblies are assumed to have equivalent reactivity. SNC has not performed a CSA for the depleted LTAs in its Vogtle, Unit 2, SFP All Cell storage configuration. Rather, the licensee is extrapolating the AOR for its existing fuel to the depleted LTAs. However, SNC has not established the appropriateness of that extrapolation. In order for the NRC staff to evaluate SNC's application against the criteria of 10 CFR 50.68(b)(4), provide the basis for extrapolating the Vogtle, Unit 2, SFP All Cell AOR to the LTAs. The All Cell CSA in the AOR should be compared to a hypothetical CSA that would otherwise be performed for the depleted LTAs for the Vogtle, Unit 2, SFP All Cell storage configuration, and the effect of any differences should be evaluated in terms of their impact on reactivity. The licensee should describe how it considered the following:

- **Regulatory Guide 1.240, "Fresh and Spent Fuel Pool Criticality," Revision 0, March 2021 (ML20356A127),**
- **NEI 12-16, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants," Revision 4 (ML19269E069), and**
- **NEI 12-16, Revision 4, Attachment C, "Appendix C: Criticality Analysis Checklist."**

<u>**Response:**</u>

In preparation for the license amendment, a full comparative analysis was performed for the LTAs for extrapolation of the analysis of record (AOR) for All Cell storage. The analysis [

]^{a,c} was then used to extrapolate the AOR to accommodate storage of the LTAs. Additional consideration has been given to References 1, 2 and Appendix C of Reference 2 (the bulleted list RAI items) and are discussed herein. Reference 2 citations within this response correspond to NEI-12-16, Revision 4, including Revision 4, Attachment C.

The following (in addition to the modeling described in Reference 3) details the LTA All Cell storage analysis approach and results.

<u>LTA All Cell Storage Analysis Approach:</u>

Storage rack dimensions for Vogtle 2 are given in Table 1. Note that the cell pitch is the "reduced" cell pitch utilized in the AOR (analysis supporting References 5 and 6) and utilized herein.

Table 1: Vogtle Unit 2 Storage Rack Geometry

Parameter	Value
Nominal Cell Pitch [inch]	10.58 N-S, 10.40 E-W
Cell Pitch [inch]	10.34 ± 0.040
Box Wall Thickness [inch]	0.075 ± 0.005
Box ID [inch]	8.75 ± 0.03
Boraflex* Width [inch]	7.75
Boraflex* Thickness [inch]	0.075
Sheathing Thickness [inch]	0.02

*Boraflex is not credited in any analysis herein.

In the analysis approach described herein, a [

]^{a,c}

- [

]a,c

[

]a,c

[

]a,c

Table 2 contains data to allow comparison of depletion input from the [ ]a,c

Table 2: Depletion Input / Design Parameter Comparison Table

a,c

Soluble boron during depletion was inferred using a maximum cycle average approach from the planned fuel management data for the LTA operation. All fuel management options considered for Cycles 24, 25 and 26 were utilized and the bounding maximum cycle average was utilized, according to the following formula.

$$\overline{C_B} = \frac{\sum_{i=1}^{Nsteps} \overline{C_B} i * \Delta BU_i}{\Delta BU_{cycle}}$$

where,

$\overline{C_B}$	=	the cycle average soluble boron concentration
$\overline{C_{B_i}}$	=	the ith burnup step average soluble boron concentration
i	=	the burnup step number
$Nsteps$	=	the total number of burnup steps in the cycle
ΔBU_i	=	the step burnup
ΔBU_{cycle}	=	the cycle burnup

While the lifetime average can also be appropriately utilized, the conservative cycle average value was applied. The candidate soluble boron concentrations (rounded up to the nearest ppm) were 734 ppm, 750 ppm, 749 ppm, 742 ppm, 732 ppm and 725 ppm. For conservatism the cycle average soluble boron concentration was rounded to a conservative value of 800 ppm.

The fuel management specific burnup profiles were selected from all potential fuel management options. The top 1, 2, 4 and 8 node burnups (on a 24 node basis) were summed, with the smallest sums identifying candidate burnup profiles. Two very similar profiles resulted from this evaluation with the chosen profile given in Table 3. A moderator temperature profile was selected by simply using the maximum temperature for each node observed for the LTAs. [

]^{a,c} The final selected profiles are given in Table 3. Note that these profiles were the conservative profiles across all operation for the LTAs, not planned discharge data.

Isotopics were output every 2 GWd up to the 75 GWd/MTU. [

]^{a,c} Additionally, to ensure modern expectations are considered, the following analytical approach was utilized concerning bias and uncertainty treatment.

Table 3: Distributed Axial Burnup and Conservative Moderator Temperature Profiles

[ ]a,c

Additional bias and uncertainty impact was determined [ ]a,c The AOR considered the following biases and uncertainties:

- 2% of TD pellet density uncertainty increase
- 0.05 wt.% uranium enrichment (increase) as an uncertainty (5.00 wt.% nominal)
- Decrease in rack cell pitch
- Decrease in rack inner dimension
- Eccentric positioning (as an uncertainty)
- Burnup uncertainty
- Methodology uncertainty
- Methodology bias
- Pool temperature bias

The AOR biases and uncertainties were considered for the LTA calculations as follows:

- Pellet Density Uncertainty Increase:
  No additional conservatism. Modeling of the LTAs is at the LTA bounding % of TD and is part of the explicit LTA impact.
- Enrichment Uncertainty:
  No additional conservatism. AOR meets modern expectations.
- Decrease in Rack Cell Pitch:
  No additional conservatism. AOR meets modern expectations.
- Decrease in Rack Inner Dimension
  No additional conservatism. AOR meets modern expectations.
- Eccentric Positioning (included as an uncertainty)
  Current expectations are to address eccentric positioning as a bias. The AOR determined a positive impact from locating the assemblies as close as possible in the storage cells. This will be acceptable, but for conservatism, the AOR eccentric positioning reactivity impact is added as a bias [
  ][a,c]
- Burnup Uncertainty
  No additional conservatism. A 5% burnup measurement uncertainty was utilized which meets modern expectations.
- Methodology Uncertainty
  No additional conservatism. The nature of the change in reactivity determination effectively cancels out the methodology bias uncertainty in the subtraction.
- Methodology Bias
  No additional conservatism. The nature of the change in reactivity determination effectively cancels out the methodology bias in the subtraction.
- Pool Temperature Bias:
  No additional conservatism. The AOR considered a range of 50 °F to 185 °F and meets modern expectations.

This addresses all included AOR items. Thus far the only additional reactivity impact was from the eccentric positioning methodology change. [

][a,c]

[

]a,c

LTA All Cell Storage Results and Conclusions:

Table 4 provides the Nominal and LTA assembly reactivity across burnup, with the burnup area of interest identified herein shaded. Table 5 provides the All Cell reactivity increase with burnup across the burnup range of interest. [

]^{a,c} For All Cell storage, the AOR burnup limit is 38979.0 MWd/MTU.

Table 4, All Cell Reactivity*	
	k_{eff}

a,c

a,c

Table 5, All Cell LTA Reactivity Increase with Burnup	
	k_{eff}

[

]a,c

a,c

Table 6 Adjusted Biases & Uncertainties for the 4oo4 Configuration	
Parameter	Impact*

[

]a,c The work described herein was put forth to eliminate any impact to storage for non-LTA assemblies. While the licensing basis has changed to accommodate the LTAs, the resulting technical specifications for non-LTA fuel remain the same.

References:

1. RG 1.240 (ML20356A127), "Fresh and Spent Fuel Pool Criticality Analyses."

2. NEI 12-16, Revision 4 (ML19269E069), "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants."

3. ML22181B156, "Vogtle Electric Generating Plant, Units 1 and 2, License Amendment Request and Exemptions to Allow Use of Lead Test Assemblies for Accident Tolerant Fuel."

4. NUREG/CR-6801 (ML031110292), "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses, U.S. Nuclear Regulatory Commission."

5. ML042320393, "Vogtle – Request to Revise Technical Specifications to Reflect Updated Spent Fuel Rack Criticality Analyses for Units 1 and 2."

6. ML052420110, "Issuance of Amendments that Revise the Spent Fuel Pool Rack Criticality Analyses, NRC to D. E. Grissette."

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

ENCLOSURE 1

SNC Response to Requests for Additional Information
Attachment 4
Affidavit

Commonwealth of Pennsylvania:
County of Butler:

(1) I, Zachary Harper, Manager, Licensing Engineering, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of Enclosure 1 to NL-22-0609 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) The information sought to be withheld is being transmitted to the Commission in confidence and, to Westinghouse's knowledge, is not available in public sources.

(iii) Westinghouse notes that a showing of substantial harm is no longer an applicable criterion for analyzing whether a document should be withheld from public disclosure. Nevertheless, public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower-case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower-case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 9/2/2022

Signed electronically by
Zachary Harper

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

ENCLOSURE 2

Revised Marked-Up TS Pages

3.7 PLANT SYSTEMS

3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

LCO 3.7.18

---NOTE---
Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4.

The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	------------------------------NOTE---------------------------- Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4. --- Verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly is in accordance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).	Prior to storing the fuel assembly in the fuel storage pool location.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries (EAB)

The VEGP site and EAB consist of approximately 3,169 acres in eastern Georgia on the west side of the Savannah River about 26 miles southeast of Augusta, Georgia, and 15 miles east-northeast of Waynesboro, Georgia, in Burke County, Georgia. The nearest point to the EAB from the VEGP Reactors is the near bank of the Savannah River. Reactor 1 is approximately 3600 feet from the EAB and Reactor 2 is approximately 3900 feet from the EAB.

4.1.2 Low Population Zone (LPZ)

The LPZ is that area falling within a 2-mile radius from the midpoint between the containment buildings.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions. In addition, LTAs 7ST1, 7ST2, 7ST3, and 7ST4, which contain fuel rods that include advanced coated cladding features, doped or standard fuel material, and up to four fuel rods with a maximum nominal U-235 enrichment of 6.0 weight percent, are permitted to be placed in limiting core regions for up to two cycles of operation without completion of representative testing.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver-indium-cadmium, or hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

f. A nominal 10.25 inch center to center pitch in the Unit 1 high density fuel storage racks.

g. LTAs 7ST1, 7ST2, 7ST3, and 7ST4 are prohibited from Unit 1 spent fuel pool storage.

(Unit 2) 4.3.1.2

--NOTE--
4.3.1.2a, 4.3.1.2d, and 4.3.1.2e do not apply to LTAs 7ST1, 7ST2, 7ST3, and 7ST4.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. K_{eff} < 1.0 when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $K_{eff} \leq 0.95$ when fully flooded with water borated to 394 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2 may be allowed unrestricted storage in the Unit 2 fuel storage pool.

e. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8 may be stored in the Unit 2 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 2 fuel storage pool in a 2-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10 may be stored

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

in the Unit 2 fuel storage pool as "low enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2. New or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments may be stored in the Unit 2 fuel storage pool as "high enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2.

Interfaces between storage configurations in the Unit 2 fuel storage pool shall be in compliance with Figures 4.3.1-3, 4.3.1-4, 4.3.1-5, and 4.3.1-6. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2. "B" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8. "C" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235. "L" assemblies are new or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10. "H" assemblies are new or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments.

f. A nominal 10.58-inch center to center pitch in the north-south direction and a nominal 10.4-inch center to center pitch in the east-west direction in the Unit 2 high density fuel storage racks.

g. For LTAs 7ST1, 7ST2, 7ST3, and 7ST4, the following requirements apply for storage in the Unit 2 spent fuel storage racks:
   1. Unrestricted storage is allowed in the 2-out-of-4 checkerboard storage configuration as shown in TS Figure 4.3.1-1.
   2. Storage is allowed in the all-cell storage configuration (“A” assemblies as shown on TS Figures 4.3.1-3 and 4.3.1-5) when the LTAs reach 64,000 MWd/MTU of burnup.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent except for LTAs 7ST1, 7ST2, 7ST3, and 7ST4 which may have four rods per assembly enriched up to 6.0 weight percent U-235;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194 foot-1 1/2 inch.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1476 fuel assemblies in the Unit 1 storage pool and no more than 2098 fuel assemblies in the Unit 2 storage pool.

Vogtle Electric Generating Plant – Units 1 and 2
Response to NRC Requests for Information
License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

ENCLOSURE 3

Revised Clean TS Pages

3.7 PLANT SYSTEMS

3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

LCO 3.7.18

---NOTE---
Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4.

The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	-----------------------------NOTE---------------------------- Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4. --- Verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly is in accordance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).	Prior to storing the fuel assembly in the fuel storage pool location.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries (EAB)

The VEGP site and EAB consist of approximately 3,169 acres in eastern Georgia on the west side of the Savannah River about 26 miles southeast of Augusta, Georgia, and 15 miles east-northeast of Waynesboro, Georgia, in Burke County, Georgia. The nearest point to the EAB from the VEGP Reactors is the near bank of the Savannah River. Reactor 1 is approximately 3600 feet from the EAB and Reactor 2 is approximately 3900 feet from the EAB.

4.1.2 Low Population Zone (LPZ)

The LPZ is that area falling within a 2-mile radius from the midpoint between the containment buildings.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions. In addition, LTAs 7ST1, 7ST2, 7ST3, and 7ST4, which contain fuel rods that include advanced coated cladding features, doped or standard fuel material, and up to four fuel rods with a maximum nominal U-235 enrichment of 6.0 weight percent, are permitted to be placed in limiting core regions for up to two cycles of operation without completion of representative testing.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver-indium-cadmium, or hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

f. A nominal 10.25 inch center to center pitch in the Unit 1 high density fuel storage racks.

g. LTAs 7ST1, 7ST2, 7ST3, and 7ST4 are prohibited from Unit 1 spent fuel pool storage.

(Unit 2) 4.3.1.2

---NOTE---
4.3.1.2a, 4.3.1.2d, and 4.3.1.2e do not apply to LTAs 7ST1, 7ST2, 7ST3, and 7ST4.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. K_{eff} < 1.0 when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $K_{eff} \leq 0.95$ when fully flooded with water borated to 394 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2 may be allowed unrestricted storage in the Unit 2 fuel storage pool.

e. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8 may be stored in the Unit 2 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 2 fuel storage pool in a 2-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10 may be stored

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

in the Unit 2 fuel storage pool as "low enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2. New or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments may be stored in the Unit 2 fuel storage pool as "high enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2.

Interfaces between storage configurations in the Unit 2 fuel storage pool shall be in compliance with Figures 4.3.1-3, 4.3.1-4, 4.3.1-5, and 4.3.1-6. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2. "B" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8. "C" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235. "L" assemblies are new or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10. "H" assemblies are new or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments.

f. A nominal 10.58-inch center to center pitch in the north-south direction and a nominal 10.4-inch center to center pitch in the east-west direction in the Unit 2 high density fuel storage racks.

g. For LTAs 7ST1, 7ST2, 7ST3, and 7ST4, the following requirements apply for storage in the Unit 2 spent fuel storage racks:
   1. Unrestricted storage is allowed in the 2-out-of-4 checkerboard storage configuration as shown in TS Figure 4.3.1-1.
   2. Storage is allowed in the all-cell storage configuration (“A” assemblies as shown on TS Figures 4.3.1-3 and 4.3.1-5) when the LTAs reach 64,000 MWd/MTU of burnup.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent except for LTAs 7ST1, 7ST2, 7ST3, and 7ST4 which may have four rods per assembly enriched up to 6.0 weight percent U-235;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194 foot-1 1/2 inch.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1476 fuel assemblies in the Unit 1 storage pool and no more than 2098 fuel assemblies in the Unit 2 storage pool.