Southern Nuclear

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

June 30, 2022

NL-22-0288
10 CFR 50.90

Docket Nos. 50-424 and 50-425

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

Southern Nuclear Operating Company
Vogtle Electric Generating Plant – Units 1 and 2

License Amendment Request and Exemptions to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

Ladies and Gentlemen:

Pursuant to the provisions of Section 50.90 and 50.12 of Title 10, Code of Federal Regulations (CFR), Southern Nuclear Operating Company (SNC) hereby requests a license amendment to Vogtle Electric Generating Plant (Vogtle) Units 1 and 2 renewed operating licenses NPF-68 and NPF-81 and exemptions to 10 CFR 50.68, 10 CFR 50.46, and 10 CFR 50 Appendix K. The proposed amendment and exemptions allow for the use of lead test assemblies (LTAs) to demonstrate operating characteristics for accident-tolerant fuel.

This amendment request proposes to add a License Condition that authorizes use of four LTAs. These LTAs will be placed in limiting core locations. In addition, exemption requests are included to allow the use of coated **AXIOM®** cladding, with **ADOPT™** fuel pellets enriched up to 6 wt% U-235. Finally, associated changes to the Operating License are requested as a result of a change to the licensing basis to 10 CFR 50.68.

On June 14, 2022, the NRC approved a one-time exception to the NRC LIC-109, LIC-101, and LIC-500 acceptance review criteria regarding Westinghouse topical reports referenced in these requested licensing actions (ML22160A686).

SNC requests approval of the proposed amendment and exemptions by July 22, 2023 to support fuel receipt for the fall 2023 Unit 2 refueling outage. The proposed changes would be implemented within 30 days after issue of the amendment.

Enclosure 1 contains information proprietary to Westinghouse Electric Company LLC (“Westinghouse”), and it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission (“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 which 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 or the supporting Westinghouse Affidavit should reference CAW-22-023 and should be addressed to Camille T. Zozula, Manager, Regulatory Compliance & Corporate Licensing.

In accordance with 10 CFR 50.91, a copy of this application is being provided to the designated Georgia official.

This letter contains no regulatory commitments. 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 the 30th day of June 2022.

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

CAG/rmj/efb/cg

Enclosure 1: *Evaluation of the Proposed Change (Proprietary)*
Enclosure 2: *Evaluation of the Proposed Change (Non-Proprietary)*
Enclosure 3: *Request for Exemption to Allow Use of AXIOM Cladding*
Enclosure 4: *Request for Exemption to Allow Use of 6 wt% Enriched Fuel Rods*
Enclosure 5: *Affidavit*

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

ENCLOSURE 2

Southern Nuclear Operating Company
Vogtle Electric Generating Plant – Units 1 and 2

License Amendment Request to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

Enclosure 2

Evaluation of the Proposed Change (Non-Proprietary)

ENCLOSURE 2

Description and Assessment of the Proposed Changes (Non-Proprietary)

ATTACHMENTS:

1. Facility Operating License Markup
2. Proposed Facility Operating License Clean Typed Pages
3. Updated Final Safety Analysis Report Markup – 10 CFR 50.68 (for information only)

1. SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for Amendment of License, Construction Permit or Early Site Permit," Southern Nuclear Operating Company (SNC) requests an amendment to Renewed Facility Operating License Numbers NPF-68 and NPF-81 for Vogtle Electric Generating Plant, Units 1 and 2. This amendment request proposes to add a License Condition to Appendix D, "Additional Conditions," of the Vogtle Unit 1 and Unit 2 Facility Operating Licenses (FOLs) that authorizes use of four Accident Tolerant Fuel (ATF) Lead Test Assemblies (LTAs) to be placed in limiting core locations for up to two cycles of operation, and provides spent fuel storage requirements and new fuel storage requirements for these LTAs.

The currently licensed fuel design and reload analysis methods do not fully accommodate the ATF LTA design and materials; therefore, the Westinghouse analytical codes and methods are supplemented as necessary, using conservative assumptions and qualitative assessments based on test results, to confirm that all applicable limits associated with the LTAs (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

In addition, discussion in the Unit 1 and Unit 2 FOLs pertaining to an exemption to 10 CFR 70.24 is removed, as both units will now rely on 10 CFR 50.68 as the licensing basis.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

The Vogtle reactors each contain 193 fuel assemblies. Each assembly consists of a matrix of 264 Zircaloy, **ZIRLO™**, or **Optimized ZIRLO®** clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, not to exceed 5 wt% enrichment. The proposed change is to load four LTAs with advanced ATF features, including ADOPT fuel [2], AXIOM cladding [3], chromium coating, and four rods per LTA with up to 6 wt% enrichment, in limiting core locations for up to two cycles of operation.

The LTAs are Westinghouse 17 X 17 **PRIME™** Optimized Fuel Assembly designs [1] and each contain (Table 1):

- Up to 132 rods with Westinghouse ADOPT uranium dioxide pellets at <5 wt% enrichment and coated AXIOM cladding
- Three rods with Westinghouse ADOPT uranium dioxide pellets at <6 wt% enrichment and coated AXIOM cladding
- One rod with Westinghouse ADOPT uranium dioxide pellets at <6 wt% enrichment and uncoated AXIOM cladding
- All other rods will have Westinghouse uranium dioxide pellets at <5 wt% enrichment, ZrB_2 IFBA coated pellets and coated AXIOM cladding

The cladding coating will consist of chromium (Cr) applied to the outer surface of the AXIOM cladding. There are no other changes to the existing fuel assembly design.

Table 1: Summary of LTA Rods

Pellet Type & Enrichment	Cladding	
	Coated AXIOM Cladding	Uncoated AXIOM Cladding
ADOPT pellets, U235 < 5 w/o	≤ 132	0
ADOPT pellets, 5 w/o ≤ U235 < 6 w/o	3	1
UO_2 with $ZrBr_2$ coating, U235 < 5 w/o	≥ 128	0
Total Rods/Assembly	263	1

2.1.1 ADOPT Fuel

Advanced Doped Pellet Technology (ADOPT) fuel is uranium dioxide containing additions of chromium and aluminum oxides.

2.1.2 AXIOM Cladding with Chromium Coating

The cladding will consist of AXIOM cladding substrate which will feature a thin layer of chromium, [ ] [a,c] results in a hard, adherent coating.

2.1.3 Higher U-235 Enrichment (6 wt%)

The four LTAs will include 16 rods (four rods per LTA) with initial enrichment of up to 6 wt%.

2.2 Current Requirements

The Vogtle Technical Specification (TS) [7] 4.2.1, "Fuel Assemblies," addresses the use of LTAs within the Vogtle reactor cores. TS 4.2.1 states, in part:

"*a limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.*"

The Vogtle TS 4.3.1, “Criticality,” and TS 3.7.18, “Fuel Assembly Storage,” limit fuel enrichment to 5 wt%.

There are currently no Operating License conditions related to LTAs; however, a 10 CFR 70.24 exemption relative to the criticality alarm portion provisions is contained within the Facility Operating License.

2.3 Reason for the Proposed Change

To achieve benefits including additional cycle length flexibility, reduced high level waste storage and disposal requirements, and positive benefits on the environmental impact of the fuel cycle (through reduced batch sizes and less waste generated), SNC has an interest in loading fuel assemblies with initial enrichment greater than 5 wt%. Loading a limited number of rods with enrichment greater than 5 wt% will ultimately provide a regulatory framework for batch loading of fuel assemblies greater than 5 wt%. In addition, these limited number of rods will provide useful data that can be used for the validation and update to the associated codes and methods.

ATF materials offer improvements, which include the ADOPT additives that facilitate greater densification and diffusion during sintering, which result in a higher density and an enlarged grain size compared to undoped uranium dioxide. While achieving the desired pellet properties, the concentration of additives has been kept at a minimum in the ADOPT design. This has the benefit of reducing the amount of parasitic neutron absorption from additives such as chromium.

AXIOM material has been developed to be more resistant to accelerated corrosion than ZIRLO or Optimized ZIRLO cladding, while meeting all fuel design criteria. As the nuclear industry pursues longer operating cycles, with increased fuel discharge burnup and fuel duty, the corrosion performance requirements for nuclear fuel cladding become more demanding. AXIOM material is being developed to be more resistant to accelerated corrosion than ZIRLO or Optimized ZIRLO cladding, while meeting all fuel design criteria. In addition, fuel rod internal pressures (resulting from the increased fuel duty, use of integral fuel burnable absorbers, and corrosion/temperature feedback effects) have become more limiting with respect to fuel rod design criteria. Reducing the associated corrosion buildup, and thus, minimizing the temperature feedback effects, provides additional margin to the fuel rod internal pressure design limit.

This ATF project will help in qualifying accident tolerant fuel for future full batch reloads. The proposed addition to the Operating License is intended to capture the impacts on the TSs needed to implement the proposed LTAs. An exemption to the requirements of 10 CFR 50.68 that restrict initial enrichment to 5 wt% is included in Enclosure 4. Attachment 3 of this enclosure shows the planned Updated Final Safety Analysis Report (UFSAR) changes as required by 10 CFR 50.68(b)(8). In addition, the placement of these LTAs in limiting locations requires an exemption to 10 CFR 50.46 and Appendix K for the use of AXIOM cladding, which is the subject of Enclosure 3.

2.4 Description of the Proposed Change

The proposed Unit 1 and Unit 2 Appendix D License Condition reads as follows:

"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 the remaining 260 fuel rods must be ≤ 5.0 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 storage 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."

In addition, a revision to the Unit 1 and Unit 2 renewed facility operating license section 2.D is proposed to remove the reference to the 10 CFR 70.24 exemption due to the proposed change in licensing basis to 10 CFR 50.68.

3. TECHNICAL EVALUATION

Introduction and Summary

This evaluation presents the technical justification supporting the conclusion that inserting the subject LTAs can be conducted in a safe manner, is bounded by the limits specified in the current analysis of record and is appropriate to support advancement of the ATF initiative.

3.1 Core Source Term

Supporting analyses for the implementation of the Vogtle lead test assembly (LTA) program include the analysis of variations in the isotopic inventory of the core. An evaluation was performed to determine the impact of 16 higher enrichment lead test rods (four LTAs, each with four higher enrichment fuel rods) on the core radionuclide inventory used for radiological/dose consequences.

The evaluation was performed using the ORIGEN-ARP sequence of SCALE. ORIGEN-ARP is a versatile point-depletion and radioactive decay computer code sequence for use in simulating nuclear fuel cycles and calculating nuclide compositions. This code sequence takes into account the transmutation of all isotopes in the material and has been widely used for tracking core inventories in commercial light water reactors.

The evaluation considered the effect over bounding ranges of enrichment (3 wt% to 6 wt% U-235), burnup (50 to 83.5 GWd/MTU), and rod power (50% to 125% of average rod power). Considering the maximum radionuclide inventory over the ranges of burnup, enrichment, and rod power considered in the evaluation, the impact of the 16 higher enrichment lead test rods on core radionuclide inventories used for radiological/dose consequences in Section 3.3 was determined to be inconsequential.

In addition to confirming that the impact of 16 higher enrichment rods is negligible, the core design implementing the LTAs was considered. A core inventory was calculated using ORIGEN ARP for the core design implementing LTAs and compared to the core inventory used for radiological/dose consequences. For significant isotopes that contribute to dose, the core inventory for the core design implementing the LTAs was determined to be bounded by the core inventory used for radiological/dose consequences.

3.2 LOCA and SLB Mass & Energy Release

The short- and long-term loss of coolant accident (LOCA) and steamline break (SLB) mass and energy (M&E) release analyses of record (AOR) were evaluated for the effects of the LTAs.

The LOCA M&E releases are affected by the following changes in fuel parameters:

- Fuel Dimensions, specifically rod outside diameter
- Pressure drops through the core
- Core Stored Energy

The long-term LOCA and SLB M&E releases can be affected by any changes in decay heat and initial reactor coolant system and steam generator conditions as a result of the LTAs. The RCS primary side initial pressure and temperatures are not changing as a result of the LTA program. The SG initial pressure, temperature and fluid volume are not changing as a result of the LTA program. These parameters are addressed below.

The short-term LOCA M&E releases are most impacted by changes in the initial RCS pressure and temperature conditions, the break location and the break area. None of these parameters are changing for the LTA program, so there is no impact on short-term LOCA M&E releases for the addition of four LTAs to the core.

The long-term LOCA M&E release analysis is performed using AOR methods. These methods model an average core, therefore the 16 individual higher enrichment rods or four LTAs are not considered separately in a core with 193 assemblies and 50,952 rods. The analysis of record which uses this methodology was compared with the updated fuel and system parameters, which included the four LTAs. None of the fuel dimensions were impacted by the LTAs and the overall core pressure drop change due to four LTAs was determined to be negligible. The core stored energy in the analysis of record was determined to be bounding for the core with the four LTAs. Finally, the decay heat curve used in the analysis of record was determined to be bounding for the four higher enriched rods present in the LTAs. Therefore, the analysis of record for the long-term LOCA M&E releases is bounding and applicable for the addition of the four LTAs. Further, there is no impact on the analysis of record for the long-term LOCA M&E releases for Vogtle Unit 1.

The SLB M&E release analyses inside and outside containment are performed using the AOR methodology. The SLB M&E release analyses model core-average parameters such as fuel heat transfer characteristics (UAs), decay heat, initial stored energy, and reactivity feedback. However, based on the total number of the fuel rods to be inserted into the core (four LTAs of 193 total fuel assemblies and up to 16 rods enriched to 6 wt% out of 50,952 total fuel rods), the impact on core-average effects such as fuel UAs, decay heat, initial core stored energy, and reactivity feedback are judged to be negligible. Therefore, the analysis of record for the SLB M&E releases inside and outside containment remain bounding and applicable with the addition of the four LTAs.

Because the LOCA and SLB M&E releases are not impacted by the LTA program, the downstream containment and compartment response analyses are also not impacted. The evaluation of the Vogtle LTAs on the short-term and long-term LOCA and long-term SLB M&E releases has determined that the analyses of record remain bounding and applicable for a core which includes the four LTAs.

3.3 Radiological/Dose Consequences

It has been determined that the LTAs do not impact the radiological consequences analyses for the following design basis accidents:

- Loss of Coolant Accident (LOCA)
- Steam Generator Tube Rupture (SGTR)
- Main Steam Line Break (MSLB)
- Loss of Offsite Power (LOOP)
- Locked Rotor (LR)
- Control Rod Ejection (CRE)
- Small Line Break Outside Containment (SLBOC)
- Waste Gas Decay Tank Rupture (WGDTR)
- Liquid Waste Tank Failure (LWTF)
- Fuel Handling Accident (FHA)

This determination is based on the following confirmations:

- The LTAs do not impact the reactor coolant system (RCS) and gas and liquid waste tank nuclide activities (SGTR, MSLB, LOOP, SLBOC, WGDTR, LWTF).
- The RCS mass released during the assumed small line break outside containment is calculated based on the assumed flow rate and is not impacted by changes in the fuel (SLBOC).
- The calculations of the steam releases from the steam generators to the environment used in the radiological consequences analyses model the core-wide fuel average temperature, the total mass in the core, and the core decay heat. None of these are impacted by the inclusion of the four LTAs (SGTR, MSLB, LOOP, LR, CRE).
- It is assumed that the LTAs lead the core and therefore could be postulated to fail following a locked rotor or rod ejection accident. It has been confirmed that inclusion of the LTAs does not increase the amount of fuel damage considered in the radiological consequences analyses of the locked rotor or rod ejection accident in the analyses of record, i.e., 5% for locked rotor with no fuel melting and 10% for rod ejection with melting limited to less than the innermost 10% of the fuel pellet at the hot spot (LR, CRE).
- It has been confirmed that the LTAs do not impact the core average nuclide activities used to determine the activity released from fuel assumed to fail following a locked rotor, rod ejection accident, or LOCA (LR, CRE, LOCA).
- It has been confirmed that the gap fractions used to define the activity released from fuel assumed to fail following a locked rotor, rod ejection, or fuel handling accident are not impacted by the differences in the LTAs from current fuel (LR, CRE, FHA). The cladding material and fuel enrichment do not impact the mechanisms of fission gas release. ADOPT fuel changes the fuel microstructure by increasing the grain size. Increased fuel grain size increases the diffusion distance of gases, resulting in lower transient fission gas release. Steady-state fission gas release is approximately the same as standard UO_2 fuel. This is consistent with the evaluation of gap release fractions in Section 6.1.1 of WCAP-18482 [2]. [

] a,c

- The activities of dose significant radionuclides postulated for release in a fuel handling accident (FHA) involving the LTAs (e.g., Xe-133, Xe-135, I-131, I-132, I-133) are bounded by the activities of the same radionuclides in the existing FHA analyses. Therefore, the dose potential of an FHA involving the LTAs is bounded by the existing fuel handling accident radiological consequence analyses when evaluated at the same decay time (FHA).

Therefore, the LTAs do not impact the radiological consequences analyses for the accidents listed above.

3.4 Non-LOCA Transients

Two categories of non-LOCA events were considered for the LTAs: those that are dependent on core-average effects and those that are impacted by local effects in the fuel rods. For the first category, events are analyzed to address gross plant criteria, such as loss of shutdown margin, margin to hot leg saturation, overpressurization of the reactor coolant system, overpressurization of the secondary system, or overfilling of the pressurizer. Based on the total number of the fuel rods to be inserted into the core (four LTAs of 193 total fuel assemblies and up to 16 higher enrichment rods out of 50,952 total fuel rods), the impact on core-average effects such as core-average fuel heat transfer characteristics, decay heat, and initial core stored energy were evaluated and determined to be negligible. Any small effects caused by the LTAs would be more than offset by existing margins in the safety analyses. As such, the LTAs do not impact the core-average events.

Events in the second category are potentially impacted by local effects in the fuel rods and could be affected more significantly by the LTAs. These events include:

- Zero and full power steamline breaks – core response cases
- Locked rotor
- Loss of reactor coolant flow (complete and partial)
- RCCA withdrawal from subcritical
- Rod ejection

Westinghouse completed an evaluation to address the potential effects of the LTAs and concluded the following:

- The LTAs have no impact on the current, approved non-LOCA computer codes, methodology, or relevant acceptance criteria for each event.
- LTA geometry, material properties, and reactivity feedback characteristics were confirmed to have no impact on the non-LOCA safety analyses. Any small effects caused by differences in the geometry, material properties, and/or reactivity feedback characteristics of the LTAs are more than offset by existing margins in the safety analyses.
- While the LTAs may lead the core, they will be placed in core locations that have been shown to be non-limiting with respect to the rod ejection analysis.
- Event-specific statepoints used as input to departure from nucleate boiling ratio (DNBR) calculations are not impacted by the LTAs.

- The relevant fuel-specific acceptance criteria continue to be met with consideration for the LTAs for events concerned with local effects: minimum DNBR (loss of flow events and RCCA withdrawal from subcritical), percent of rods in DNB and peak clad temperature (locked rotor), and peak fuel enthalpy (rod ejection).

In summary, the LTAs have been evaluated against the non-LOCA safety analyses and were determined to be acceptable. All acceptance criteria are met and the conclusions documented in the applicable UFSAR sections remain valid.

3.5 Small Break and Large Break LOCAs

Vogtle is currently licensed with the 1981 Westinghouse Large Break Loss-of-Coolant Accident (LOCA) Evaluation Model (EM) using BASH [15] and the 1985 Westinghouse Small Break LOCA Evaluation Model using NOTRUMP [16, 17]. Both evaluation models have been developed to meet the requirements presented in Appendix K to 10 CFR 50.

To support the insertion of the four LTAs, Large Break LOCA and Small Break LOCA evaluations were performed to justify safe operation of the LTAs and to estimate the impact of the LTAs on the co-resident fuel. The Large Break and Small Break LOCA evaluations described in this section are valid up to the licensed burnup limit.

Westinghouse reviewed the BASH EM and NOTRUMP EM to assess the impact of the LTA features and determined that the approved codes and methods are adequate to evaluate the LTAs without any modification. The Large Break and Small Break evaluations demonstrate that the acceptance criteria for LOCAs, given in 10 CFR 50.46, continue to be met. [

] a,c

For Large Break LOCA, the main design aspects of the LTA fuel were evaluated for operation up to the current licensed burnup limit using the approved BASH EM via select calculations and qualitative assessments. The core-wide thermal-hydraulic response during the blowdown and reflood portions of the Large Break LOCA transient are negligibly impacted by the presence of four LTAs and therefore the impact of the LTA is based on the cladding heatup response determined by the LOCBART code. For the LTAs, the resulting peak cladding temperature (PCT) is bounded by the current co-resident fuel, the maximum local oxidation is less than 17%, and the core-wide oxidation is less than 1.0%. [

] a,c Lastly, the co-resident fuel is negligibly impacted by the presence of the LTAs and therefore, continues to meet all 10 CFR 50.46 acceptance criteria.

Two limitations and conditions are imposed by the Nuclear Regulatory Commission (NRC) as part of the approval of BASH EM [19]:

1. Future usage of BASH EM will be limited to (a) assessments pursuant to the reporting requirements of 10 CFR 50.46; and (b) evaluations to support minor

plant, fuel design, or other input changes that would normally be handled under 10 CFR 50.46 and/or 10 CFR 50.59.

2. BASH EM shall not be used for any future Large Break LOCA evaluations for changes that would be expected to significantly exacerbate downcomer boiling (for example, closure of the residual heat removal discharge crosstie valves, early initiation of the recirculation sprays, a significant increase in downcomer metal heat capacity).

The Large Break LOCA evaluation has been conducted pursuant to the requirements of 10 CFR 50.46 and utilizes the approved methods [15, 19]; the resulting estimated impact on PCT is reported accordingly. Downcomer boiling is not exacerbated since the core-wide thermal-hydraulic response is negligibly impacted due to the presence of four LTAs. Based on these considerations, this evaluation complies with the conditions and limitations imposed by the NRC in the final safety evaluation [19] and use of the BASH EM as described herein is acceptable.

For Small Break LOCA, the LTA fuel design was evaluated for operation up to the current licensed burnup limit using the approved NOTRUMP EM via qualitative assessments. A Small Break LOCA transient is relatively slow progressing and is characterized by a quasi-stratified top-down draining of the reactor coolant system. The stored energy of the fuel is removed prior to core uncovery in a Small Break LOCA transient, and additionally, Vogtle has a low beginning-of-life-limited PCT. As such, the Small Break LOCA evaluation concluded that the existing Small Break LOCA analysis is representative of the LTAs. [

] a,c The co-resident fuel is negligibly impacted by the presence of the LTAs and therefore, continues to meet all 10 CFR 50.46 acceptance criteria.

The Large Break and Small Break LOCA evaluations have considered the impact of installing four LTAs at Vogtle and operating the LTAs up to the current licensed burnup limit. It is concluded that the existing BASH EM and NOTRUMP EM analyses-of-record for Vogtle are representative of the LTAs and the presence of the LTAs will have a negligible impact on the co-resident fuel. Therefore, the 10 CFR 50.46 acceptance criteria continue to be met. [
] a,c For the purposes of 10 CFR 50.46 reporting, the impact of the LTAs are estimated as a 0°F change in PCT. The cycle-specific fuel reload evaluations will ensure that acceptance criteria are met for the insertion of LTAs.

3.6 Design Transients

The core reactivity parameters were reviewed, and it was determined that any differences caused by the inclusion of the LTAs have a negligible impact on the margin to trip and control systems operability analyses. The results and conclusions of the analysis of record remain valid for the LTA program.

3.7 Fuel Rod Design

In general, the impact of fuel rod lead use materials is beneficial for fuel performance. These features include:

- AXIOM Fuel Cladding Material
- Fuel Rod Chromium (Cr) Coating

- ADOPT Fuel Pellets

The fuel performance features of AXIOM cladding are documented in [3], which is under NRC review. AXIOM cladding has demonstrated better in-reactor corrosion performance compared to Optimized ZIRLO Fuel Cladding Material, especially in high duty operating environments. Fuel rod Cr-coating also provides improved corrosion resistance to the cladding; however, no corrosion benefits are taken for the fuel performance evaluations of the Vogtle LTA program. For the LTA program, the fuel rod Cr-coating is assumed to have the same material properties and behaviors as the substrate material (AXIOM cladding), with no credit taken for additional corrosion benefits. The chromium coating is modeled as an increase in the outer diameter of the cladding as part of the fuel performance analyses, and neutronic penalties are accounted for indirectly as part of the neutronics input to the fuel rod design (FRD) analyses.

The fuel performance features of ADOPT fuel are documented in [2], for which the NRC has issued a final safety evaluation [33]. ADOPT fuel is a modified uranium dioxide (UO_2) pellet doped with small amounts of chromia (Cr_2O_3) and alumina (Al_2O_3). The additives facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2.

Fuel performance calculations for the Vogtle LTAs consider the effects of the new fuel products using the latest set of fuel performance models, PAD5 [9]. When necessary, changes are made to the PAD5 models and methods to analyze the new LTA fuel features. For AXIOM cladding and ADOPT fuel, these changes are consistent with the as-submitted topical reports [2, 3, 9] and all subsequent NRC requests for additional information (RAIs). No corrosion resistance credit is taken for the Cr-coating, as discussed previously.

Some rods in the LTA assembly are intended to exceed the current licensed fuel rod initial U-235 enrichment (5 wt%) limit. Although not approved beyond this limit in the NRC Safety Evaluation (SE) [9], the PAD5 fuel performance models were developed to consider operation beyond 6 wt% U-235 enrichment. PAD5 is used to perform the fuel rod design evaluations for any rod which exceeds 5 wt% enrichment.

The design limits are confirmed using the latest fuel performance models, including those currently under review with the NRC [2, 3], as part of the standard reload analysis performed for each cycle.

3.8 Nuclear Design

Current standard 17x17 VANTAGE+ design dimensions are employed for the fuel rods and LTAs. Integral Fuel Burnable Absorbers (IFBA) are not used in the ADOPT fuel rods.

All parameters associated with the fuel pellets and rods are modeled conservatively. No adverse core physics impacts are anticipated from the proposed activity. The neutronically significant features of the chromium coating and ADOPT fuel pellets (including those above 5 wt% U-235) are explicitly modeled; however, the coating and the ADOPT pellet additives have a negligible neutronic impact. There is no change to the standard overall nuclear design process in terms of incore fuel management, safety analyses, or operational data evaluation.

The current methods licensed for Vogtle, PARAGON [22] and NEXUS qualification [32] are used to neutronically model the core including the LTAs. Given the small number of fuel rods with pellet enrichment exceeding 5 wt% U-235 within each LTA, the LTA neutron flux spectrum is established by the balance of rods with enrichment less than 5 wt% such that the resulting neutron flux spectrum is similar to the currently operating core. The effect of the fuel rods with pellets enriched above 5 wt% U-235 is confined to the intra-assembly power distribution.

The performance of the current methods is benchmarked using the PARAGON2 lattice code [13] which was approved for fuel enrichments beyond 5 wt% U-235. The benchmark is performed to ensure that the pin power reconstruction is not biased and that the peaking factor uncertainties (applied in the analysis of peaking factors and fuel melting, and during Technical Specification surveillance) remain conservative.

The LTAs are expected to be leading the core in peaking factors during the first cycle and the beginning of the second cycle; however, the LTA peaking factors which also account for mixed core effects are less limiting than those assumed in the plant UFSAR.

Online core monitoring with the **BEACON™** Core Monitoring System (i.e., the Power Distribution Monitoring System) will not be affected by the LTAs, and the ability to accurately calculate the reactor 3-dimensional power shape will not be affected. The small number of LTA fuel rods with enrichment above 5 wt% are placed such as to have a negligible effect on the incore flux detectors. Technical Specification Surveillance Requirements are not impacted, and design basis peaking factor limits will be met at all times.

The features in the LTA that are different from the co-resident fuel assemblies have no effect on the moderator temperature coefficient (which is a global core reactivity parameter) or on the validity of the conditional exemption of end-of-life MTC measurement.

3.9 Thermal-Hydraulic Design

As described in Section 4.4 of the Vogtle Units 1 & 2 UFSAR [8], the Thermal-Hydraulic (T/H) design methodology applied to the plant Departure from Nucleate Boiling (DNB) analyses consists of DNB correlations such as WRB-2 [24], the Westinghouse version of the VIPRE-01 subchannel code, referred to as the VIPRE-W code [25], and the Revised Thermal Design Procedure (RTDP) [26] for determination of a 95/95 DNB Ratio (DNBR) limit. The rod bow evaluation methodology is described in [27] and [28]. The transition core evaluation method is described in [29]. No modification or update is required to any of the NRC-approved topical reports on the T/H design methodology in the UFSAR for application to the LTAs.

The ADOPT fuel pellet does not affect the fuel cladding DNB performance as determined from DNB experiments in the DNB correlation database. The AXIOM cladding material does not change any fuel rod geometric parameters or characteristics that could adversely affect DNB performance as compared to the Optimized ZIRLO cladding. There is no change in the DNB correlations and the VIPRE-W modeling method in the UFSAR for the LTA fuel rods containing ADOPT fuel pellets with the AXIOM cladding material.

The T/H design methods for the chromium coated cladding evaluation were reviewed in accordance with the NRC interim guidance [10]. 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 chromium coated cladding increases the resistance to oxidation and surface wear. The DNB performance of the coated fuel rods is similar to that of the uncoated fuel rods. The similarity in the DNB performance is confirmed through comparative tests between coated and uncoated tubes conducted at the Westinghouse Advanced Loop Tester (WALT) loop [11] and the University of Wisconsin testing loop [21]. The chromium coated cladding tubes are designed and manufactured to meet the current fuel design specifications on surface roughness for friction loss. The chromium coating thickness is controlled so that there is no significant change in the flow area of the fuel assembly containing coated fuel rods, since a reduction in the flow area of the LTA could result in an increase in the pressure drop and a mixed core DNBR penalty. As the LTAs are designed to have a lower power peaking factor than that used in the T/H analysis of record for the UFSAR, sufficient DNBR margin is available to offset the potential mixed core penalty on the LTAs. There is no change in the UFSAR Section 4.4 and T/H input to the plant Technical Specifications and UFSAR Chapter 15 DNB analyses.

The VIPRE-W code [25] can perform steady-state and transient DNBR calculations and non-LOCA post-Critical Heat Flux (CHF) fuel rod transient analysis based on the fuel design input, including the fuel temperatures, applicable to the LTA. The method using the VIPRE-W code for the DNB propagation evaluation is described in [30]. The cladding burst model applicable to the AXIOM cladding as discussed in [3] is input to the DNB propagation evaluation. There is no change in the acceptance criteria and conditions of the DNB propagation evaluation method in [30] for the LTA evaluation. Fuel failure due to DNB and DNB propagation are not expected to occur in the LTA fuel rods during a Condition III or IV non-LOCA event, when the reload evaluation indicates that the DNBR values remain above the design limit DNBR.

3.10 Fuel Assembly Design

3.10.1 Materials

The cladding will consist of AXIOM cladding substrate which will feature a thin layer of chromium, [ ] a,c results in a hard, adherent coating. The resulting microstructure is dense [ ] a,c

The specified [ ] a,c of uncoated AXIOM cladding.

[ ]a,c Balloon and burst testing of [ ]a,c cladding demonstrated [ ] a,c burst temperature and [ ] a,c balloon strain [ ]a,cGiven that [ ] a,c

The high temperature oxidation [ ] a,c of AXIOM cladding [ ] a,c up to a peak cladding temperature (PCT) of [ ] a,c High temperature testing of chromium coated [ ] a,c compared to uncoated cladding. The addition of a [ ] a,c chromium coating has [ ] a,c uncoated cladding, and [ ] a,c of coated AXIOM [ ] a,c

Testing of chromium coated [ ] a,c Testing of [ ]a,c chromium coated AXIOM at [ ]a,c for [ ] a,c compared to uncoated AXIOM. Because chromium coating [

]a,c

The corrosion rate is determined by the outer surface material of the fuel rods, and [ ]a,c the normal operation corrosion of [ ]a,c chromium coated AXIOM cladding [

]a,c

[

]a,c

3.10.2 Mechanical Design

Westinghouse will evaluate the mechanical design impact of the Lead Test Assemblies and their subcomponents. These items will include the fuel rod interface/interaction with the top nozzle, bottom nozzle, hold-down springs, guide thimble and instrument tube, grid assembly, and joints and connections. The chromium-coated clad results in a slight change to the cladding outer diameter. The use of ADOPT pellets has a minimal impact on rod and assembly weight based on the small increase in fuel density with ADOPT [2].

No component changes or changes to basic fuel assembly design requirements are expected, and no adverse mechanical design impacts are anticipated from the proposed activity.

The interface between the LTAs and the 17x17 VANTAGE+ with Debris Mitigation Features fuel design with PRIME features will be assessed to ensure no changes in spacer grid or fuel rod support system are required. No grid-to-rod fretting or grid damage is anticipated. There is no change to the LTA interface with any other plant equipment, and there is no change to any fuel handling tools, equipment, or procedures. No impact is anticipated on the lost parts analysis. There will not be any change or impact to the storage of the LTAs as the LTA weight is minimally changed. The LTA shipping and handling loads will be evaluated and documented; no adverse impact is expected.

The fuel rod mechanical design is based on the 17x17 Westinghouse Optimized Fuel Assemblies (OFA) fuel rod. All standard fuel rod design criteria will be evaluated for the fuel rods to ensure that sufficient margin exists to any rod failure or damage criterion. This evaluation includes all fuel rod performance requirements, heat transfer requirements, fuel boundary integrity requirements, fuel rod internal pressure requirements, requirements for fuel rod support and positioning, and plenum spring design criteria. The debris fretting resistance of the coated fuel rods is similar to that of the standard rods.

Fabrication of the fuel rods will be performed using standard techniques. For all rods except those with pellets above 5 wt% U-235 enrichments, pellet inspection, rod loading and characterization, and welding of the coated clad rods will be performed at the commercial Columbia Fuel Fabrication Facility. The fuel rods with pellets above 5 wt% U-235 enrichment will be fabricated and loaded into the LTAs at [ ]a,c All required rod inspections will be performed using standard or augmented inspection techniques, including X-ray, UT, calibrated gauge, and leak check.

A fuel examination work scope has been planned to confirm the expected performance of the LTAs and fuel rods. This proposed post-irradiation examination plan includes high magnification visual exams, rod cleaning, profilometry, oxide thickness measurement, and eventual shipment of fuel rods to a hot cell for destructive evaluation.

3.10.3 Seismic

Any explicit analyses evaluating LTAs featuring AXIOM cladded fuel rods coated by [ ]^{a,c} PRIME features, and ADOPT pellets are expected to be bounded by previous analyses. Additional analyses have shown that existing dynamic models remain applicable for fuel with chromium coated cladding, PRIME features, and ADOPT pellets. Therefore, grid impact results remain applicable for the core with LTAs.

3.11 RCS Chemistry

Any increase in the RCS activity, caused by fuel oxidation arising from the introduction of reactor coolant into the fuel rod during normal operation, is detected and monitored by existing plant equipment in accordance with approved procedures (i.e., no changes to the RCS radiochemistry procedures will be needed). There are no significant new fuel reliability concerns anticipated; it is projected that the fuel rod will perform well in all modes of operation; and no adverse interactions with the current RCS chemistry regime are anticipated.

The formation and possible release of Cr-51 is an issue that is monitored through ongoing surveillance at the plant. The process is already in place to evaluate the radioisotopes and the gaseous and liquid effluents and to report this information to the NRC on an annual basis. If Cr-51 in the coolant begins to challenge plant dose release limits, it will be observed to increase as more of the fuel in the core is transitioned to Cr-coated cladding. In this case, systems can be implemented to effectively remove this radioisotope before it becomes a safety problem. Similarly, with the impact of Cr ions on the coolant chemistry, a surveillance plan put in place alongside the implementation of Cr-coated cladding to monitor the coolant chemistry will mitigate any impact of Cr ions. The impact of fast neutron irradiation on Cr mechanical properties is inherently included in material property correlations and limits that are developed based on irradiated material as described in ATF-ISG-2020-01 [10].

3.12 Criticality

Introduction

The following sections detail the methodology and results for the Vogtle High Burnup Higher Enrichment (HBHE) Spent Fuel Pool (SFP) criticality analysis for storage of LTAs within the Vogtle New Fuel Storage Racks (NFSRs) and Unit 2 SFP.

Computer Codes

The analysis methodology employs the following computer codes and cross-section libraries: (1) the two-dimensional (2-D) transport lattice code PARAGON Version 1.2.0 [22] and its cross-section library based on Evaluated Nuclear Data File Version VI.3 (ENDF/B-VI.3), and (2) Scale Version 6.2.3, as documented in [23], with the ENDF/B-VII 238-group cross-section library.

All pool and new fuel storage rack reactivity calculations are performed with the CSAS5 module of Scale 6.2.3 with the ENDF B-VII 238 group cross section library, using the CENTRM cross section processing method for lattice cell/multiregion cell treatment.

Storage of the Vogtle HBHE LTAs within the Unit 2 SFP and the NFSRs was evaluated with consideration of the current AOR for the Unit 2 SFP and the NFSRs.

LTA Assembly and Assembly Model Characteristics:

The HBHE LTAs consists of four 17x17 Westinghouse OFAs with advanced features. The following parameters and modeling assumptions are important assembly characteristics concerning spent fuel pool criticality and are a mix of final design input, current fuel management expectations and conservative assumptions.

- 260 ~4.95 wt% U-235 enriched fuel rods (uncertainty to 5 wt%)
- Four ~5.95 wt% U-235 enriched fuel rods (assumed at 6 wt% nominal)
- 128 IFBA rods with 1.5X standard loading (^{10}B reduced 5% were applicable in SFP models only) and 8” cutback region (IFBA modeled as full length in depletion analysis)
- IFBA rods contain annular blankets but are modeled as solid rods.
- Non-IFBA rods are ADOPT doped pellets: ADOPT doped pellets can be bounded by a maximum fuel percent of Theoretical Density (TD) of UO_2 of 98.3% and are modeled as such without dopants.
- All rods except one higher enriched rod are Chromium coated AXIOM cladding, while the additional rod is uncoated AXIOM cladding. SFP and NFSR models contain uncoated zircaloy-4 (neutronically equivalent to AXIOM cladding) or AXIOM cladding. Depletion analysis input applied a 10 μm chromium coating.
- 16 Wet Annular Burnable Absorber (WABA) rodlets modeled conservatively as full length in the depletion analysis to 24 GWd/MTU.
- Non-reactive assembly structures like mixing and spacer grids, sleeves, and top/bottom nozzles are not modeled.

Analysis and Results

<u>New Fuel Storage Racks</u>

Design input for the NFSRs was obtained from the intended LTA fuel characteristics and the AOR. The NFSR was modeled as seen in the planar (radial) layout in Figure 1 except where described herein. Planar reflective boundary conditions are applied on all sides. Green in Figure 1 represents a 12” thick concrete wall on the sides and bottom (24” gap to the top of the model and 24” gap from the bottom of the fuel assemblies to the concrete floor after considering reflection). Additional details for a nominal storage cell are in Figure 2, with an axial view in Figure 3. The center-to-center distance for assemblies across the larger intermodule gap in the typical “Y” dimension is 51.75 inches. The concrete wall is conservatively modeled up against the NFSR arrays, with calculations indicating this is conservative for fully flooded and optimum moderation conditions compared to models with the planar gap that exists to the concrete wall.

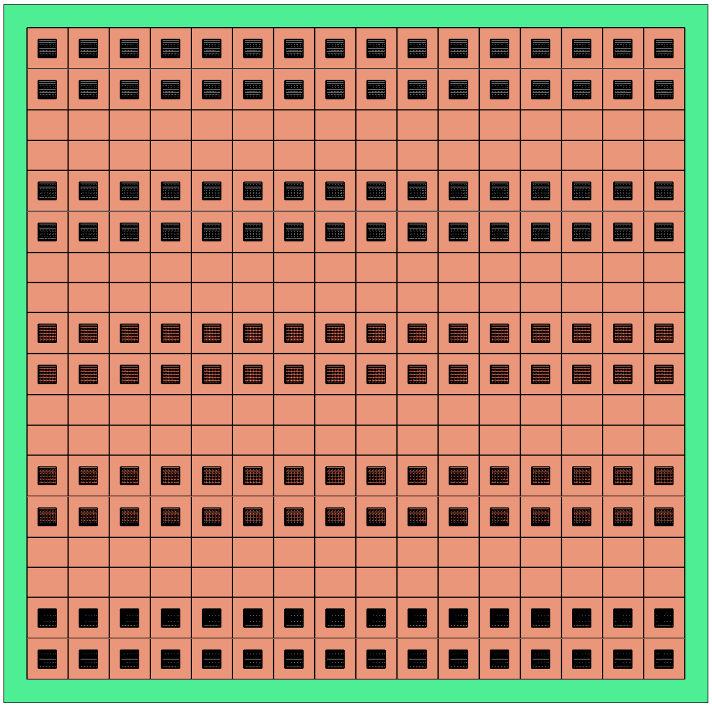

Figure 1: Vogtle New Fuel Rack Model Radial Layout

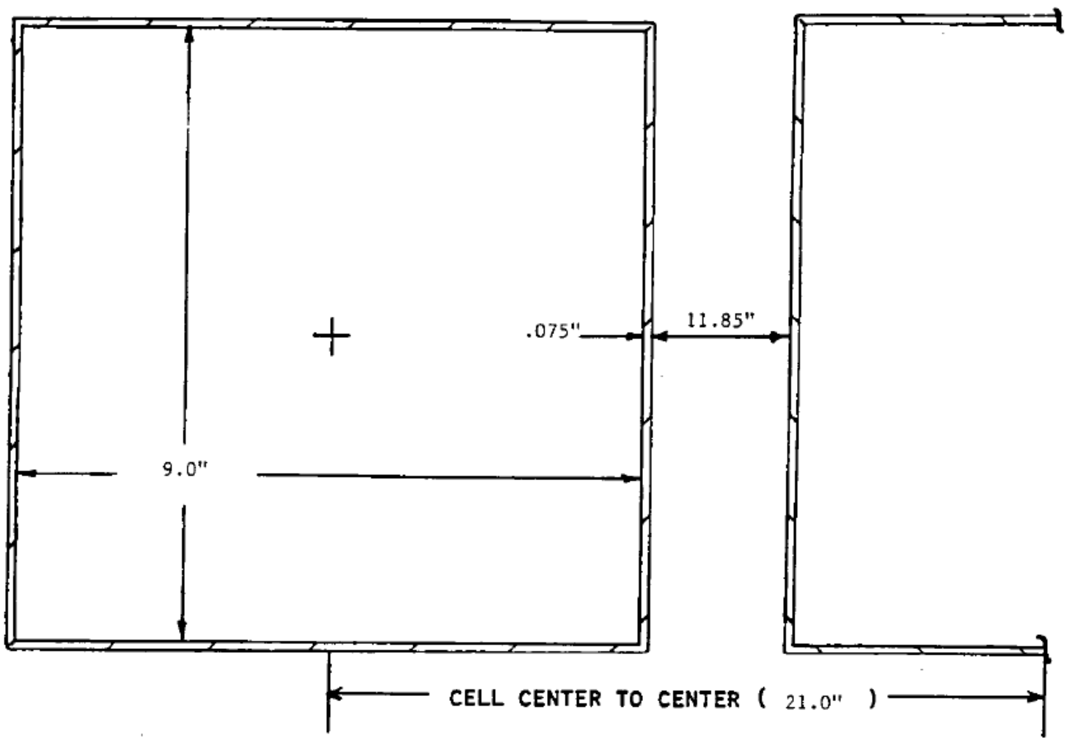

Figure 2: Vogtle New Fuel Storage Cell Nominal Dimensions

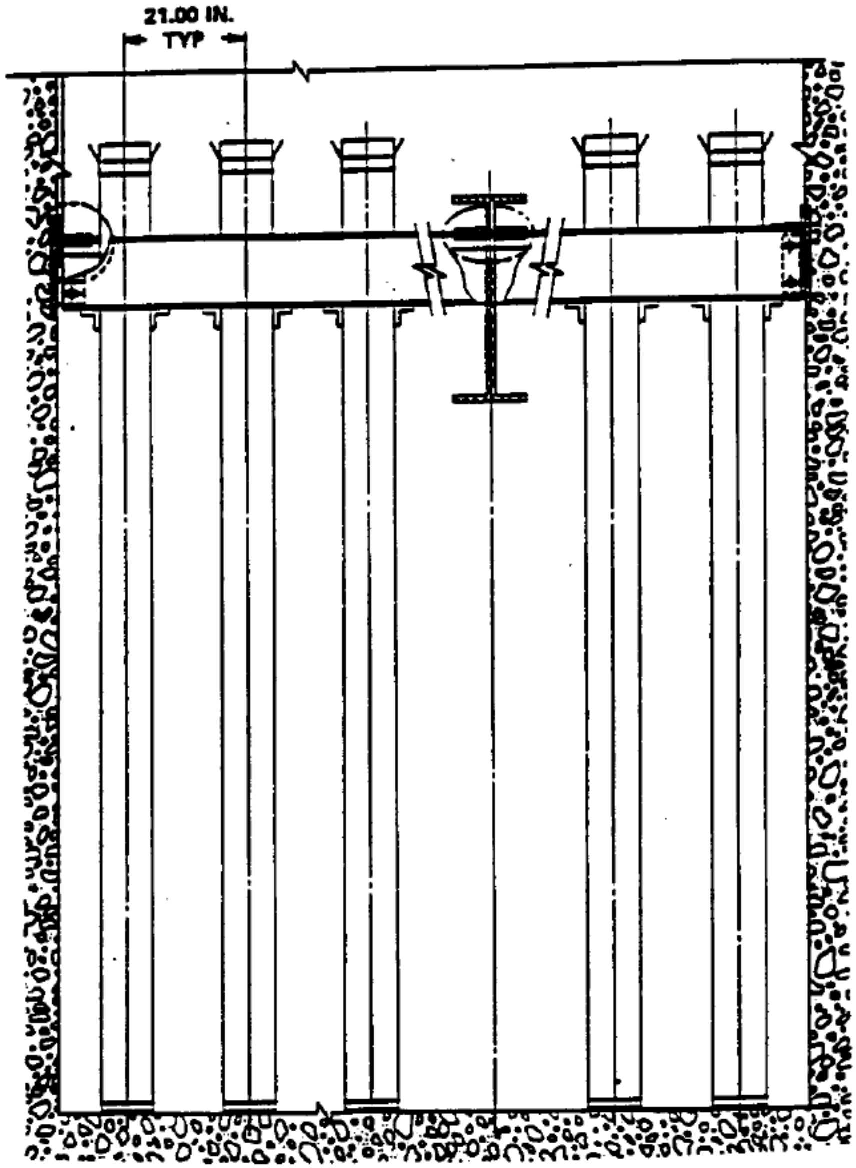

Figure 3: Vogtle New Fuel Rack Axial Layout

The NFSRs were modeled under dry, fully flooded and optimum moderation conditions for 5 wt.% fuel (bounding current design) and LTA assemblies. Results of the NFSR modeling are given in Table 2.

Table 2: NFSR Reactivity Results – Current vs LTA Assemblies[1]

	Current Fuel Assembly (5 wt% no IFBA)	LTA Fuel Assembly (128 IFBA)
Dry	0.7539[2]	0.8276[2,3]
Fully Flooded	0.9219[2]	0.8037[2]
Optimum Moderation (0.03 g/cc water density)	0.9575	0.8554

[1]Maximum Monte Carlo Uncertainty of 0.00007 from all cases.
[2]Determined from a model representing a single infinitely reflected storage cell.
[3]All fuel rods at 6 wt% bounding enrichment with no IFBA, to show higher enrichment fuel will not exceed dry reactivity limits.

A full updated rack-up of biases and uncertainties should be on the order of that for currently licensed fuel which results in acceptable storage reactivity. Crediting the installed IFBA, Table 2 shows significant reactivity hold down is present to preclude any need for a detailed bias and uncertainty analysis. The intended LTAs additionally contain WABA which is not credited. Thus, it is safe to load the LTAs unrestricted in the NFSRs.

<u>Spent Fuel Pool Storage</u>

Storage in the spent fuel pool was explicitly evaluated for the two-out-of-four (2oo4), and all-cell (4oo4) storage configurations from the SFP AOR for Unit 2. Storage racks are modeled as in the current analysis of record with a reduced cell pitch model.

2oo4 (no burnup credit) Storage Configuration

The 2oo4 storage configuration was modeled with and without 128 IFBA rods and does not specifically credit any installed WABA. The 2oo4 model with reference 5.0 wt% fuel and no IFBA produced a calculated k_{eff} of 0.9455. The same 2oo4 configuration model with the LTA assemblies with 128 IFBA rods yielded a k_{eff} of 0.7970. Thus, the LTAs have a reactivity margin of about 15% at fresh conditions. Peak reactivity will rise early in the assembly life as IFBA burns out but will not challenge the 15% margin. As a result, sufficient reactivity hold-down is present to conclude it is safe to load the LTAs in the 2oo4 SFP storage configuration.

All Cell (including burnup credit) Storage Configuration

For storage of the LTAs in the all-cell configuration, a burnup limit of 64 GWd/MTU was selected. The AOR burnup requirement is about 40 GWd/MTU and a 64 GWd/MTU LTA burnup limit provides a 24 GWd/MTU or greater than 8% in k_{eff} margin for the LTAs. At 64 GWd/MTU, no additional analysis is needed to allow safe storage of the LTAs in the all cell storage configuration.

Accident Conditions, Interfaces and Other Considerations

The limiting accident in the SFP is a multiple misload [14]. An infinitely modeled multiple misload (4 ATF LTA assemblies in a 2x2 reflected storage array) with TS-required 2000 ppm of soluble boron results in total reactivity of 0.9469, including a total bias and uncertainty of 0.045, bounding all analysis of record bias and uncertainty totals. This reactivity is without any IFBA (which was shown to provide significant hold-down) or WABA within the model. Additionally, the analysis considered an infinite misload of LTAs when only four will be operated. As a result, the LTAs do not create an accident condition concern.

The SFP soluble boron credit concentration is not specifically calculated for the LTAs. Given the significant conservatisms present in all storage scenarios evaluated, the current normal condition soluble boron concentration remains applicable. No interface analysis is updated and with the LTAs conservatively residing in acceptable storage configurations, all analysis of record interface analysis remains applicable.

Upon final discharge, the LTAs may undergo reconstitution, with a fuel rod or rods removed for testing. This action must take place in an isolated area of the spent fuel pool such that no assembly is face adjacent to the LTA being reconstituted. Only one fuel rod at a time may be removed. A stainless-steel rod must replace the fuel rod before any other fuel rod is removed, thus limiting any reactivity impact from rod removal.

SFP and NFSR Analysis Summary and Conclusions

Storage of the LTAs was assessed in the NFSRs, 2oo4 and all-cell storage configurations. Results concluded fresh LTAs can be placed unrestricted within the NFSRs and the 2oo4 SFP storage configuration. For storage in the SFP all-cell configuration, it is concluded that 64 GWd/MTU of burnup is needed prior to storage within.

3.13 Probabilistic Risk Assessment Impacts

The following four PRA parameters could be impacted due to the placement of LTAs in the core:

- Decay heat level at the time of reactor trip due to initiating events
- The hottest core node temperature
- Core Exit Thermocouple temperature
- Unfavorable Exposure Time in Anticipated Transient Without Scram (ATWS)

Based on a review of the details of the LTA fuel loading, it is estimated that there would be very small increase (<0.01%) in the core averaged decay heat generation level. The small increase in the decay heat level would result in a negligible impact on the T/H analyses results using the Modular Accident Analysis Program (MAAP) software. Therefore, the impact of LTA on decay heat level at the time of reactor trip is not anticipated to impact the success criteria, the event timings, and the core damage sequences in the current Vogtle PRA model.

It was estimated that, following reactor trip, the LTA assembly would be less than 0.6% hotter than the previous assembly at that location. This small increase would have a negligible to minor impact on the core peak temperature response. The negligible to minor impact of the core peak temperature response is not anticipated to impact the success criteria, the event timings, and the core damage sequences in the current Vogtle PRA model in any significant way to increase Vogtle PRA risks.

Core exit thermocouple temperature is used to monitor the core cooling critical safety function status to enter functional recovery emergency operating procedures and severe accident management guidance. The assessments of LTA bundle power indicate that the LTA assembly power increase would be in the order of 1.006. The local assembly temperature increase is estimated to be approximately 5° F, which is in the order of the CET measurement uncertainty. There are many core exit thermocouples at Vogtle 2. The plasma display shows minimum, average, and maximum core exit thermocouple temperatures for each reactor quadrant. Since the plasma display needs inputs from all core exit thermocouples throughout all 4 reactor quadrants, approximately 5 degrees F increase in the local assembly temperature, which is on the order of the core exit thermocouple measurement uncertainty, would not affect the core exit thermocouple temperature displayed on the plasma display in the main control room in any significant way to affect event timings and operator actions related to entering proper Emergency Operating Procedures (EOPs) or Severe Accident Management Guidelines (SAMGs) and taking recovery actions. Therefore, PRA risks would be minimally affected.

The unfavorable exposure time durations are one of the most important variables in determining core damage in the ATWS PRA model. An unfavorable exposure time is the duration of time in a cycle during which pressure relief by pressurizer PORVs and safety valves is not sufficient to prevent RCS pressure from exceeding stress level C limit during the initial pressure transient after ATWS. It is estimated that the LTA would not impact the system transient portion of the ATWS analysis or the MTC limit over 95% of the cycle. Therefore, it is judged that the LTAs would not impact the ATWS events. As such, there is also no impact on unfavorable exposure time. Hence, it is judged that there would be no impact on PRA risks from ATWS sequences where core damage occurs due to the unfavorable exposure time.

Since the PRA risks are negligibly impacted by the LTA as mentioned above, the impact on Surveillance Frequency Control Program or the Risk-Informed Completion Time is anticipated to have minimal impact as well.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

- GDC 10 Reactor design – Requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 11 Reactor inherent protection – States that the reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

- GDC 12 Suppression of reactor power oscillations – Requires the reactor core and associated coolant, control, and protection systems to be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 19 Control room – Requires a control room to be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

- GDC 27 Combined reactivity control systems capability - The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- GDC 28 Reactivity limits - The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase.

- GDC 35 Emergency core cooling - Section 50.46 provides a means (via analytical requirements and prescriptive analytical limits) to satisfy General Design Criterion 35, “Emergency core cooling,” in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic licensing of production and utilization facilities.”

- GDC 61 Fuel storage and handling and radioactivity control - The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

- GDC 62 Prevention of criticality in fuel storage and handling - Requires the licensee to limit the potential for criticality in the fuel handling and storage system by physical systems or processes. This requirement will continue to be met as discussed in the Vogtle Updated Final Safety Analysis Report for the receipt, handling, and storage of the LTAs.

4.2 Precedent

For this ATF license amendment request, precedents exist for many of the individual features of the project, including the use of ADOPT fuel, AXIOM cladding, chromium coating on a zirconium-based cladding, and LTAs in limiting core positions.

4.2.1 Use of ADOPT

ADOPT fuel has been used previously in LTAs. The most recent example is for the LTA program at Byron Unit 2 [12].

4.2.2 Use of AXIOM

AXIOM cladding has been used at a number of earlier LTA programs, including VC Summer [4], Millstone Unit 3 [5], and Byron Units 1&2 [6].

4.2.3 Use of Chromium Coating

Chromium coating on a zirconium-based cladding has been used previously in LTAs. The most recent example is for the LTA program at Byron Unit 2 [12].

4.2.4 Use of LTAs in Limiting Core Positions

The placement of LTAs in limiting core locations has been approved for the LTA program at Byron Unit 2 [12], although this application was clarified to be for steady state only.

4.2.5 Use of FOL Condition to apply alternate requirements from TS

The use of an FOL Condition to offer alternate requirements from TS has been used previously. The majority of plants in the industry have adopted TSTF-448-A Rev. 3, *Control Room Habitability.* Many of these submittals included a FOL Condition to allow Surveillance Requirements (SRs) verifying the assessment of CRE habitability and the measurement of CRE pressure to be considered met without the associated SRs being performed within the specified frequency (plus the 1.25 times allowance provided by SR 3.0.2).

In addition, many Improved Technical Specification (ITS) conversion amendments provide alternate requirements for performing SRs that are new or revised. (For a recent example, see the Sequoyah Nuclear Plant ITS Issuance of Amendments, pkg ML15238B499.)

4.3 No Significant Hazards Consideration Determination Analysis

Overview

In accordance with 10 CFR 50.90, "Application for Amendment of License, Construction Permit or Early Site Permit," Southern Nuclear Operating Company (SNC) requests an amendment to Renewed Facility Operating License (FOL) Nos. NPF-68 and NPF-81 for Vogtle Electric Generating Plant (Vogtle) Units 1 and 2. This amendment request proposes to add a License Condition to Appendix D, "Additional Conditions," of the Vogtle Unit 1 and Unit 2 Operating License that authorizes use of four Lead Test Assemblies (LTAs) to be placed in limiting core locations. In addition, discussion in the Unit 1 and Unit 2 FOLs pertaining to an exemption to 10 CFR 70.24 is removed, as both units will now rely on 10 CFR 50.68 as the licensing basis.

The currently licensed fuel design and reload analysis methods do not fully accommodate the LTA design and materials; therefore, the Westinghouse analytical codes and methods are supplemented as necessary using conservative assumptions and qualitative assessments based on test results, to confirm that all applicable limits associated with the LTAs (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

According to 10 CFR 50.92, "Issuance of Amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
(3) Involve a significant reduction in a margin of safety.

SNC has evaluated the proposed change for Vogtle, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

Criteria

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?** No.

The proposed change involves a small number of LTAs, which are conservatively designed from a neutronic standpoint, and are thermal-hydraulically and mechanically compatible with all plant Systems, Structures and Components (SSCs). The fuel pellets and fuel rods themselves will have no impact on accident initiators or precursors. The use of a small number of fuel rods enriched to 6 wt% has a negligible impact on analytical results, and the analyses of record remain bounding. There will not be a significant impact on the operation of any plant SSC or on the progression of any operational transient or design basis accident. There will be no impact on any procedure or administrative control designed to prevent or mitigate any accident.

The Westinghouse LTAs are of the same design as the co-resident fuel in the core, with the exception of AXIOM cladding (with and without chromium-coated cladding) and ADOPT fuel, containing a limited number of higher enriched fuel rods in place of the standard fuel rods. The LTAs will be placed in limiting core locations; however, the reload designs will meet all applicable design criteria. Evaluations of the LTAs will be performed as part of the cycle specific reload safety analysis to confirm that the acceptance criteria of the existing safety analyses will continue to be met. Operation of the Westinghouse coated AXIOM and ADOPT fuel will not increase the predicted radiological consequences of accidents currently postulated in the Updated Final Safety Analysis Report.

Further, the small increase in U-235 enrichment in the four LTAs has been conservatively evaluated. Placement of these LTAs within the new and spent fuel storage racks is restricted within the assumptions of the evaluation to ensure a criticality event does not occur.

Removing discussion in the Unit 1 and Unit 2 FOLs pertaining to an exemption to 10 CFR 70.24 has no impact on the probability or consequences of an accident previously evaluated.

Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?** No.

The proposed change involves the use of a small number of LTAs which are very similar in all aspects to the co-resident fuel. The proposed change does not change the design function or operation of any SSC, and does not introduce any new failure mechanism, malfunction, or accident initiator not considered in the current design and licensing bases.

The reactor cores will be designed to meet all applicable design and licensing basis criteria. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident.

The reload core designs for the cycles in which the Westinghouse LTAs will operate will demonstrate that the use of the LTAs in limiting core locations is acceptable. The relevant design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of Westinghouse LTAs does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

Further, the small increase in U-235 enrichment in the four LTAs has been conservatively evaluated. Placement of these LTAs within the new and spent fuel storage racks is restricted within the assumptions of the evaluation to ensure a criticality event does not occur.

Removing discussion in the Unit 1 and Unit 2 FOLs pertaining to an exemption to 10 CFR 70.24 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed change will not create the possibility of a new or different kind of accident than those previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?** No.

Operation with four Westinghouse LTAs, placed in limiting core locations, does not change the performance requirements on any system or component such that any design criteria will be exceeded. The current limits on core operation defined in the Vogtle Technical Specifications will remain applicable to the subject LTAs during the two cycles of operation. Westinghouse analytical codes and methods are used, and supplemented as necessary using conservative assumptions, to confirm that all applicable limits associated with the LTAs (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

With respect to non-fuel SSCs, there is no reduction in the margin of safety for any safety limit, limiting safety system setting, limiting condition of operation, instrument setpoint, or any other design parameter.

The storage restrictions placed on the slightly enriched fuel rods in the LTAs ensures the margin of safety is not significantly reduced.

Removing discussion in the Unit 1 and Unit 2 FOLs pertaining to an exemption to 10 CFR 70.24 does not involve a reduction in the margin of safety.

Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the evaluation presented above, there is high confidence that utilization of four LTAs containing a limited number of Westinghouse AXIOM and ADOPT (with and without chromium-coated cladding) accident tolerant fuel rods for two cycles of operation will have a negligible impact on any aspect of reactor operations or reactor safety. Westinghouse analytical codes and methods are supplemented as necessary using conservative assumptions and qualitative assessments based on test results, to confirm that all applicable limits associated with the LTAs (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the site licensing basis and Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

SNC has evaluated this proposed operating license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Assessments." SNC has determined that these proposed changes to use four Lead Test Assemblies (LTAs) containing a limited number of Westinghouse AXIOM-cladded slightly enriched fuel rods and ADOPT accident tolerant fuel at Vogtle meet the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for Categorical Exclusion; Identification of Licensing and Regulatory Actions Eligible for Categorical Exclusion or Otherwise Not Requiring Environmental Review," and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of Amendment." This determination is based on the fact that these changes are being proposed as an amendment to the license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.3, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change does not result in an increase in power level, does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts. It is expected that all plant equipment would operate as designed in the event of an accident to minimize the potential for any leakage of radioactive effluents. The proposed changes will have a negligible impact on the amounts of radiological effluents released offsite during normal at-power operations or during accident scenarios.

Previous NRC analysis of the environmental impacts associated with the shipment of spent fuel were limited to an enrichment of up 5% (assumed to be weight percent) with the peak rod to current approved levels of 62,000 MWd/MTU (megawatt-days per metric ton of uranium). The four rods with an enrichment up to 6% in each LTA has a negligible impact on the overall assembly enrichment (<0.02% increase) and the burnup remains less than the current licensing basis. Further evaluation of the environmental impacts associated with the transport and post irradiation examination of the irradiated fuel rods would be required as part of the qualification requirements for the cask system used to transport for testing in the future.

Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There is no change in individual or cumulative occupational radiation exposure due to the proposed change. Specifically, the proposed change to use four slightly enriched LTAs containing a limited amount of Westinghouse accident tolerant fuel with AXIOM and ADOPT fuel pellets for two cycles of operation has no impact on any radiation monitoring system setpoints. The proposed action will not change the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed action result in any change in the normal radiation levels within the plant.

Therefore, in accordance with 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment need be prepared in support of the proposed amendment.

6. REFERENCES

[1] LTR-NRC-22-7, Revision 0, Fuel Criterion Evaluation Process Notification of the 17X17 OFA PRIME Fuel Product Implementation, February 28, 2022. (ML22059B071)

[2] WCAP-18482, Revision 0, Westinghouse Advanced Doped Pellet Technology (ADOPT) Fuel, May 2020. (ML20132A015)

[3] WCAP-18546, Revision 0, Westinghouse AXIOM Cladding for Use in Pressurized Water Reactor Fuel, March 2021. (ML21090A110)

[4] VC Summer Nuclear Station – Exemption from the Requirements of 10 CFR Part 50, Sections 50.44, 50.46, and Appendix K, January 2005. (ML05004029)

[5] Millstone Power Station Unit 3 – Proposed Exemption Request for the Use of AXIOM Cladding Material in Lead Test Assemblies (LTAs), June 2016. (ML16189A104)

[6] Byron Station, Unit 1 and 2 – Exemption from the Requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR50, Appendix K, June 2006. (ML061380518)

[7] Vogtle Electric Generating Plant Units 1 and 2 Technical Specifications, Amendments 214/197.

[8] Vogtle Electric Generating Plant Updated Final Safety Analysis Report, Revision 24, August 2021.

[9] WCAP-17642, Revision 1, Westinghouse Performance Analysis and Design Model (PAD5), May 2016. (ML17090A443)

[10] ATF-ISG-2020-01, Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept, January 2020. (ML19343A121)

[11] Guoqiang Wang, William A. Byers, Zeses Karoutas, “Baseline WALT DNB Test Results with Cr Coated Cladding to Support Accident Tolerant Fuel Development,” Proceedings of the ASME 2021 28th International Conference on Nuclear Engineering, ICONE-28-66591, August 4-6, 2021.

[12] Byron Station Unit 2 – License Amendment Request (LAR) to Utilize Accident Tolerant Fuel Lead Test Assemblies, March 2018. (ML19038A017)

[13] WCAP-18443, Revision 0, Qualification of the Two-Dimensional Transport Code PARAGON2, October 2019. (ML21214A335)

[14] NEI 12-16, Revision 4, Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor (LWR) Power Plants, September 2019. (ML19269E069)

[15] WCAP-10266-P-A, Revision 2, “The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code,” March 1987. (TER only ML20213F318)

[16] WCAP-10079-P-A, Revision 0, “NOTRUMP A Nodal Transient Small Break and General Network Code,” August 1985. (TER only ML20125C624)

[17] WCAP-10054-P-A, Revision 0, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” August 1985. (TER only ML20125C624)

[18] U.S. NRC Regulatory Guide 1.223, “Determining Post Quench Ductility,” (Draft), 2016 (ML16005A134).

[19] WCAP-10266-P-A, Revision 2, Addendum 3-A, Revision 1, “Incorporation of the LOCBART Transient Extension Method into the 1981 Westinghouse Large Break LOCA Evaluation Model with BASH (BASH-EM),” October 2007. (TER only: ML20213F318)

[20] U.S. NRC Regulatory Guide 1.224, “Establishing Analytical Limits for Zirconium-Alloy Cladding Material,” (Draft), 2016 (ML16005A133).

[21] Zeses E. Karoutas, et al, “Evaluation of Coated Cladding for Increased Resistance to Fuel Rod Failure due to DNB during PWR Flow Reduction Transients,” The 19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-19), 35695, March 6-11, 2022.

[22] WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON,” March 2003, ML030760103

[23] ORNL/TM-2005/39, “Scale: A Modular Code System for Performing Standard Computer Analyses for Licensing Evaluation,” November 2006

[24] WCAP-10444-P-A, “Reference Core Report VANTAGE 5 Fuel Assembly”, Westinghouse Electric Corporation, September 1985, ML20126K599

[25] WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” Westinghouse Electric Company LLC, October 1999, ML993160158

[26] WCAP-11397-P-A, “Revised Thermal Design Procedure,” Westinghouse Electric Corporation, April 1989, ML080630437

[27] WCAP-8691-R1, Revision 1, "Fuel Rod Bow Evaluation," Westinghouse Electric Corporation, July 1979, TER only: ML19282A022

[28] LTR-NRC-11-40, "Re-send of WCAP-8691-R1, 'Fuel Rod Bow Evaluation' and Accompanying SER for Information Only (Proprietary)," Westinghouse Electric Company, August 2011, ML20079K326

[29] WCAP-11837-P-A, "Extension of Methodology for Calculating Transition Core DNBR Penalties," Westinghouse Electric Corporation, January 1990, ML20154J560

[30] WCAP-8963-P-A, Addendum 1-A, Revision 1-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)," Westinghouse Electric Company LLC, June 2006, ML052290092

[31] WCAP-13749-P-A, Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement, ML20117B946

[32] WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology", July 2020, ML20191A256

[33] Final Safety Evaluation for WCAP-18482-P/WCAP-18482-NP, Revision 0, Westinghouse Advanced Doped Pellet Technology (ADOPT) Fuel, June 13, 2022, ML2212A156

ATTACHMENT 1

Proposed Facility Operating License Markup for Units 1 and 2

Vogtle Electric Generating Plant Unit 1
Renewed Operating License No. NPF-68

2.C.(11) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. XXX, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

Appendix D

Amendment Number	Additional Condition	Implementation Date
XXX	***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 the remaining 260 fuel rods must be ≤ 5.0 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 storage 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.***	***Within 30 days of the issuance of the amendment.***

Vogtle Electric Generating Plant Unit 1
Renewed Operating License No. NPF-68

2.D.

~~The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 5.~~

~~An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1967, issued August 21, 1986, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.~~

~~These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items b and c above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.~~

The facility requires an exemption from the requirements of paragraph III.D.2(b) (ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 5. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

Vogtle Electric Generating Plant Unit 2
Renewed Operating License No. NPF-81

2.C.(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. XXX, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

Appendix D

Amendment Number	Additional Condition	Implementation Date
XXX	***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 the remaining 260 fuel rods must be ≤ 5.0 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 storage 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.***	***Within 30 days of the issuance of the amendment.***

Vogtle Electric Generating Plant Unit 2
Renewed Operating License No. NPF-81

2.D.

~~The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 8.~~

~~An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1981, issued July 13, 1988, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.~~

~~These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemption in item b above is granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.~~

The facility requires an exemption from the requirements of paragraph III.D.2(b) (ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 8. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

ATTACHMENT 2

Proposed Facility Operating License Clean Typed Pages

<table>
<tr><th>Amendment Number</th><th>Additional Condition</th><th>Implementation Date</th></tr>
<tr><td>___</td><td>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 the remaining 260 fuel rods must be ≤ 5.0 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 storage 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.</td><td>Within 30 days of the issuance of the amendment.</td></tr>
</table>

7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(11) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. ___, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires an exemption from the requirements of paragraph III.D.2(b) (ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 5. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 162, as supplemented by a change approved by License Amendment No. 175.

F. GPC shall comply with the antitrust conditions delineated in Appendix C to this license.

Amendment Number	Additional Condition	Implementation Date
179	Southern Nuclear Operating Company (SNC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the licensee amendments No. 173 (Unit 1) and No. 155 (Unit 2). SNC is approved to utilize the seismic probabilistic risk assessment (SPRA) model for use in the categorization process rather than the previously approved seismic margin approach. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.	Within 90 days of the issuance of the amendment.
___	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 the remaining 260 fuel rods must be ≤ 5.0 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 storage 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.	Within 30 days of the issuance of the amendment.

Amendment Number	Additional Condition	Implementation Date
___ Continued	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.	Within 90 days of the issuance of the amendment.

successfully demonstrated prior to the time and condition specified below for each:

a) DELETED

b) DELETED

c) SR 3.8.1.20 shall be successfully demonstrated at the first regularly scheduled performance after implementation of this license amendment.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. ___, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires an exemption from the requirements of paragraph III.D.2(b) (ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 8. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

   Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 144, as supplemented by a change approved by License Amendment No. 175.

F. GPC shall comply with the antitrust conditions delineated in Appendix C to this license.

G. Southern Nuclear shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as approved in the SER (NUREG-1137) through Supplement 9 subject to the following provision:

   > Southern Nuclear may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. Deleted.

I. The Owners shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

ATTACHMENT 3

UFSAR Markup – 10 CFR 50.68

The following introductory paragraph will be added to UFSAR Section 4.3.2.6.1:

“Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and fuel storage facilities and by administrative control procedures in accordance with 10 CFR 50.68(b). This section identifies those criteria important to criticality safety analyses.”

Southern Nuclear Operating Company
Vogtle Electric Generating Plant – Units 1 and 2

License Amendment Request to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

Enclosure 3

Request for Exemption to Allow Use of AXIOM Cladding

Request for Exemption to Allow Use of AXIOM Cladding

1.0 PURPOSE

Pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 50.12, "Specific exemptions," Southern Nuclear Operating Company (SNC) requests an exemption from the provisions of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," for the Vogtle Electric Generating Plant (VEGP). The requested exemption would permit the use of AXIOM fuel rod cladding material in lead test assembly (LTA) applications. The regulations in 10 CFR 50.46 contain acceptance criteria for the emergency core cooling system (ECCS) for reactors that have fuel rods fabricated either with zircaloy or ZIRLO fuel rod cladding material. Concurrently, 10 CFR 50, Appendix K, Section I.A.5, requires the Baker-Just equation be used to calculate the rate of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction in the core. The Baker-Just equation assumes the use of a zirconium alloy other than AXIOM material.

Therefore, an exemption is required from specific portions of both 10 CFR 50.46 and 10 CFR 50, Appendix K to support the use of AXIOM fuel rod cladding in a limited number of LTAs (7ST1, 7ST2, 7ST3, and 7ST4) at VEGP. This exemption request relates solely to the specific cladding material identified in these regulations (fuel rods with zircaloy or ZIRLO cladding) and will provide for the application of 10 CFR 50.46 and 10 CFR 50, Appendix K acceptance criteria to LTA designs utilizing AXIOM fuel rod cladding at VEGP.

2.0 BACKGROUND

As the nuclear industry pursues longer operating cycles, with increased fuel discharge burnup and fuel duty, the corrosion performance requirements for nuclear fuel cladding become more demanding. AXIOM material was developed to be more resistant to accelerated corrosion than ZIRLO or Optimized ZIRLO cladding, while meeting all fuel design criteria. In addition, fuel rod internal pressures (resulting from the increased fuel duty, use of integral fuel burnable absorbers, and corrosion/temperature feedback effects) have become more limiting with respect to fuel rod design criteria. Reducing the associated corrosion buildup, and thus, minimizing the temperature feedback effects, provides additional margin to the fuel rod internal pressure design limit. (Note that an exemption was granted to 10 CFR 50.46 and 10 CFR 50 Appendix K [10], which allowed the 10 CFR 50.46 acceptance criteria and the Baker-Just equation to be applied to fuel assembly designs using the Optimized ZIRLO fuel rod cladding material at VEGP.)

AXIOM cladding variants have been included in past Lead Test Rod (LTR) programs for Virgil C. Summer Nuclear Station [1 and 2] and Byron Nuclear Power Plants [3 and 4] that included lead rod average burnups up to 75,000 MWD/MTU. A final AXIOM alloy composition was selected based on the results observed in these LTR programs and included in an LTA program for Millstone Unit 3. The

NRC approved the Millstone Unit 3 exemption from 10 CFR 50.46 and Appendix K [5].

3.0 TECHNICAL JUSTIFICATION OF ACCEPTABILITY

Westinghouse topical report WCAP-18546-P, "Westinghouse AXIOM Cladding for Use in Pressurized Water Reactor Fuel," [6] provides the details and results of tests for AXIOM cladding along with the material properties proposed for use in various models and methodologies when analyzing AXIOM fuel cladding, including the use of 1985 Westinghouse Small Break Loss-of-Coolant Accident (LOCA) Evaluation Model with NOTRUMP [7 and 8]. A review of the 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code [9] concludes the existing models and correlations, which are the same or similar to those discussed in [6] for the NOTRUMP evaluation model, are acceptable to assess AXIOM cladding for the VEGP LTAs. Section 3.5 of Enclosures 1 and 2 of the license amendment request attendant to this exemption describes the VEGP Loss-of-Coolant Accident (LOCA) evaluation performed for the LTAs with AXIOM cladding.

SNC has proposed a License Condition for VEGP to reflect that the LTAs will be placed in limiting locations for up to two cycles of operation. The details of these limiting locations and the assessment to allow the use of AXIOM fuel rod cladding for LTA application is provided in Enclosures 1 and 2 of the license amendment request attendant to this exemption. Future reload evaluations will ensure that acceptance criteria are met for the insertion of LTAs composed of fuel rods with AXIOM cladding.

4.0 JUSTIFICATION OF EXEMPTION

10 CFR 50.12, "Specific exemptions," states that the Commission may grant exemptions from the requirements of the regulations of this part provided two criteria are met. These criteria are: (1) the exemption authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; and (2) the Commission will not consider granting an exemption unless special circumstances are present. The requested exemption to allow the use of AXIOM fuel rod cladding material in addition to zircaloy or ZIRLO at VEGP satisfies these criteria as described below.

<u>Criterion 1</u>

a. This exemption is authorized by law. The selection of a specified cladding material in 10 CFR 50.46, and implied in 10 CFR 50, Appendix K, was adopted at the discretion of the Commission consistent with its statutory authority. No statute required the NRC to adopt this specification. Additionally, the NRC has the authority under 10 CFR 50.12 to grant exemptions from the requirements of Part 50 upon showing proper justification. SNC is not seeking an exemption from the acceptance and analytical criteria of 10 CFR 50.46 and 10 CFR 50, Appendix K. The intent of this request is solely to allow the use of criteria set forth in these regulations for application to the AXIOM fuel rod cladding material.

b. This exemption will not present an undue risk to public health and safety. Reload evaluations ensure that acceptance criteria are met for the insertion of LTAs with fuel rods clad with AXIOM material. Due to similarities in the composition of the AXIOM alloy and the Optimized ZIRLO and standard ZIRLO alloys, fuel assemblies using AXIOM fuel rod cladding are evaluated using plant-specific models to address the changes in the cladding material properties. The LOCA safety analyses for VEGP are supported by the applicable site-specific Technical Specifications (TS). Reload cores are required to be operated in accordance with the operating limits specified in the TS. Thus, the granting of this exemption request will not pose an undue risk to public health and safety.

c. This exemption is consistent with the common defense and security. As noted above, this exemption request is only to allow the application of the aforementioned regulations to allow testing of an improved fuel rod cladding material. All the requirements and acceptance criteria will be maintained. The special nuclear material in these assemblies is required to be handled and controlled in accordance with approved procedures. Use of LTAs with AXIOM fuel rod cladding will not affect plant operations and is consistent with common defense and security.

Criterion 2

Special circumstances support the issuance of an exemption. 10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. This exemption request meets the special circumstance criteria of 10 CFR 50.12(a)(2)(ii), “Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.” For VEGP, application of the subject regulations is not necessary to achieve the underlying purpose of the rule.

10 CFR 50.46 identifies acceptance criteria for ECCS performance at nuclear power plants. Westinghouse will perform an evaluation using LOCA methods as described in Enclosure 1 of this submittal to ensure that assemblies with AXIOM fuel rod cladding material meet all LOCA safety criteria.

The intent of 10 CFR Part 50, Appendix K, paragraph I.A.5 is to apply an equation that conservatively bounds for rates of energy release, hydrogen generation, and cladding oxidation from a metal-water reaction (i.e., the Baker-Just equation). Due to the similarities in the composition of the AXIOM alloy and the Optimized ZIRLO and standard ZIRLO fuel rod cladding materials, application of the Baker-Just equation is anticipated to be applicable for the AXIOM alloy.

5.0 CONCLUSION

The 10 CFR 50.46 and 10 CFR 50, Appendix K regulations are currently limited in applicability to the use of fuel rods with zircaloy or ZIRLO cladding. 10 CFR 50.46 and 10 CFR 50, Appendix K do not apply to the proposed use of AXIOM fuel rod cladding material because AXIOM has a slightly different composition than zircaloy or ZIRLO. With the approval of this exemption request, these regulations will be applied to AXIOM fuel rod cladding in LTA applications at VEGP.

In order to support the use of AXIOM fuel rod cladding material in LTA applications at VEGP, an exemption from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K is requested. Pursuant to 10 CFR 50.12, the requested exemption is authorized by law, does not present undue risk to public health and safety, and is consistent with the common defense and security. Approval of this exemption request does not violate the underlying purpose of the rule. In addition, special circumstances exist to justify the approval of an exemption from the subject requirements.

6.0 REFERENCES

[1] Letter to Jeffery B. Archie (SCE&G) from Karen R. Cotton (NRC), "V. C. Summer Nuclear Station – Exemption from the Requirements of 10 CFR Part 50, Sections 50.44, 50.46, and Appendix K (TAC No. MC4462)," January 14, 2005. (ML050040249)

[2] Letter to Jeffrey B. Archie (SCE&G) from Timothy J. McGinty (NRC), "Virgil C. Summer Nuclear Station, Unit 1 – Correction to Exemptions Issued on March 13, 2008, from the Requirements of 10 CFR Part 50, Sections 50.44, 50.46, and Appendix K (TAC No. MD5699)," May 23, 2008. (ML081220827)

[3] Letter to Christopher M. Crane (Exelon) from Robert F. Kuntz (NRC), "Byron Station, Unit Nos. 1 and 2 – Exemption from the Requirements of 10 CFR 50.44, 10 CFR 50.46, and 10 CFR Part 50, Appendix K (TAC Nos. MC8517 and MC8518)," June 30, 2006. (ML061380518)

[4] Letter to Charles G. Pardee (Exelon) from Marshall J. David (NRC), "Byron Station, Unit No. 2 – Exemption from the Requirements of 10 CFR Part 50, Section 50.46 (TAC No. MD8455)," April 30, 2009. (ML090490645)

[5] Letter to Daniel G. Stoddard (DNC) from Richard V. Guzman (NRC), "Millstone Power Station, Unit No. 3 – Exemption from the Requirements of 10 CFR 50.46 and Appendix K of 10 CFR Part 50, to Allow the Use of AXIOM Cladding Material in Lead Test Assemblies (CAC No. MF8210)," May 10, 2017. (ML17087A308)

[6] Licensing Package issued to U.S. NRC Document Control Desk from Korey L. Hosack (Westinghouse), "Submittal of Westinghouse Topical Report WCAP-18546-P / NP, "Westinghouse AXIOM Cladding for Use in Pressurized Water Reactor Fuel"," March 31, 2021. (ML21090A110)

[7] Westinghouse Topical Report WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985. (ML20125C624 (includes [8]))

[8] Westinghouse Topical Report WCAP-10079-P-A, "NOTRUMP A Nodal Transient Small Break and General Network Code," August 1985.

[9] Westinghouse Topical Report WCAP-10266-P-A, Revision 2, “The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code,” March 1987. (ML20213F318)

[10] “Southern Nuclear Operating Company Exemption from NRC Requirements: Use of Optimized ZIRLO Fuel Rod Cladding Material for Joseph M. Farley Nuclear Plant, Units 1 and 2, and Vogtle Electric Generating Plant, Units 1 and 2,” August 4, 2016. (ML16179A410)

Southern Nuclear Operating Company
Vogtle Electric Generating Plant – Units 1 and 2

License Amendment Request to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel

Enclosure 4

Request for Exemption to Allow Use of 6 wt% Enriched Fuel Rods

Request for Exemption from 10 CFR 50.68(b)(7) to Allow Use of 6 wt% Enriched Fuel Rods

1.0 PURPOSE

Pursuant to Title 10 of the *Code of Federal Regulations* (CFR) 50.12, "Specific exemptions," Southern Nuclear Operating Company (SNC) requests an exemption from provision (b)(7) of 10 CFR 50.68 “Criticality accident requirements" for the Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The requested exemption would permit the use of four Lead Tests Assemblies (LTA) with a limited number of fuel rods with a maximum nominal U-235 enrichment of up to six percent by weight for VEGP operation. This exemption request relates solely to the limited number of fuel rods in four LTAs as detailed in the amendment request provided in Enclosure 1 to this submittal.

The regulations in 10 CFR 50.68, specifically section (b)(7), states that “The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.”

Therefore, an exemption is required from 10 CFR 50.68(b)(7) to support the use of the four LTAs with a maximum nominal U-235 enrichment of six percent by weight.

2.0 BACKGROUND

SNC is pursuing a limited number of fuel assemblies with a limited number of rods containing fuel with a slight increase in enrichment. An Operating License amendment for VEGP Units 1 and 2 is required to allow the new LTA fuel assemblies to be stored in the new fuel storage racks (NFSRs) and the spent fuel pool (SFP). The amendment request is provided in Enclosure 1 to this submittal.

An exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area was previously granted with NRC materials license No. SNM-1967, which exempted Georgia Power Company (GPC) and Southern Nuclear from the criticality alarm system provision of 10 CFR 70.24 insofar as that section applies to the storage of fuel assemblies held under this license. The Vogtle Units 1 and 2 licensing basis meets all the other 10 CFR 50.68 criticality accident requirements.

The proposed exemption from 10 CFR 50.68(b)(7) will allow for receipt, inspection, and storage of the LTAs prior to loading the LTAs into the VEGP reactor vessel. The proposed exemption would also apply to storage of the LTAs in the SFP after they have been removed from the reactor vessel.

3.0 TECHNICAL JUSTIFICATION OF ACCEPTABILITY

The LTAs will be shipped to the Vogtle site in Westinghouse Traveller-B STD containers which have been approved by NRC Certificate of Compliance for radioactive material packages number 9380 [1].

The Westinghouse Traveller shipping containers were approved for transporting Type A or Type B quantities of fissile radioactive material in the form of new (unirradiated) PWR fuel Assemblies, with a maximum allowable enrichment for uranium dioxide (UO_2) fuels to 6 weight percent U-235 for PWR fuel assemblies. The criticality analysis considered the addition of Group 4 fuel assemblies as allowable contents for the Traveller shipping container. Group 4 fuel assemblies consist of zirconium clad rods with UO_2 pellets enriched up to a maximum of 6 weight percent U-235. The cladding may include a chromium coating or an Optimized ZIRLO Liner (OZL), and the UO_2 pellets may consist of ADOPT fuel material. The NRC concluded that the Traveller shipping container, containing Group 4 fuel assemblies, will meet the criticality safety requirements of 10 CFR Part 71 [1].

The LTAs may be temporarily stored in their shipping containers prior to placement in their designated storage locations: the new fuel storage racks (NFSRs) and the SFP storage racks. The criticality analyses for the SFP, NFSRs, and fuel handling equipment were evaluated for handling and storage of the LTAs. Impacts to criticality for dry cask storage of the LTAs will be analyzed in the future. Per [1], no more than 20 loaded shipping containers will be temporarily stored at one time in the New Fuel Shipping Container Laydown and Unloading areas. Upon removal of each fuel assembly from its shipping container, it is inspected and surveyed for external contamination. The fuel assembly is then transferred to its designated storage location as specified by a fuel movement procedure. Criticality safety in the Container Laydown and Unloading areas and storage locations is maintained by limiting interaction between adjacent fuel assemblies. In addition, the design of the storage locations, combined with plant procedures, will ensure that the possibility of accidental criticality during fuel handling and storage activities is remote. Therefore, the need for criticality monitors will continue to be precluded during the receipt, handling, and temporary storage of the LTAs.

Enclosure 1 to this submittal provides the details and results of evaluations showing that the LTAs can be safely stored within the NFSRs and the SFP in accordance with 10 CFR 50.68, with the exemption request to 10 CFR 50.68(b)(7) documented herein. References to 10 CFR 50.68 acceptability in this section assume the exemption request herein.

For the NFSRs, standalone analysis confirmed acceptability of unrestrictive LTA storage in the NFSRs, meeting the requirements outlined in 10 CFR 50.68. For the spent fuel pool, LTA storage analysis was evaluated for the 2-out-of-4 and all-cell configurations.

The storage conditions are addressed with a proposed license condition to the Vogtle Units 1 and 2 Operating License in Enclosure 1 to this submittal. Accident and interface impacts were considered without impact on the conclusions of the analysis of record for the storage configurations outlined herein.

4.0 JUSTIFICATION OF EXEMPTION

10 CFR 50.12, "Specific exemptions," states that the Commission may grant exemptions from the requirements of the regulations of this part provided two criteria are met. These criteria are: (1) the exemption authorized by law, will not

present an undue risk to the public health and safety, and is consistent with the common defense and security; and (2) the Commission will not consider granting an exemption unless special circumstances are present. The requested exemption for the receipt, storage, and handling of four LTAs with a limited number of fuel rods with a maximum nominal U-235 enrichment of six percent by weight satisfies these criteria as described below.

Criterion 1

a. This exemption is authorized by law. The selection of a specified enrichment in 10 CFR 50.68(b) was adopted at the discretion of the Commission consistent with its statutory authority. No statute required the NRC to adopt this specification. Additionally, the NRC has the authority under 10 CFR 50.12 to grant exemptions from the requirements of Part 50 upon showing proper justification. The intent of this request is solely to allow a limited quantity (16 rods total in 4 LTAs) to exceed the limit specified in 50.68(b)(7).
b. This exemption will not present an undue risk to public health and safety. The analysis performed in support of the enclosed LAR addresses the safe storage of the LTAs. These were evaluated using AOR plant-specific models to determine that existing storage configurations can be used for storage of the LTAs, with restrictions (see Section 3 above and Section 3.12 of Enclosure 1). Thus, the granting of this exemption request will not pose an undue risk to public health and safety.
c. This exemption is consistent with the common defense and security. As noted above, this exemption request is only to allow the application of the aforementioned regulations to a limited number of fuel rods with a maximum nominal U-235 enrichment of 6 weight percent. The special nuclear material in these assemblies is required to be handled and controlled in accordance with approved procedures. Possession of the LTAs containing 4 fuel rods each with a maximum nominal U-235 enrichment of 6 weight percent at VEGP will not affect plant operations and is consistent with common defense and security.

Criterion 2

Special circumstances support the issuance of an exemption.

10 CFR 50.12(a)(2) states that the Commission will not consider granting an exemption to the regulations unless special circumstances are present. This exemption request meets the special circumstances criteria of 10 CFR 50.12(a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." For VEGP, application of the subject regulations is not necessary to achieve the underlying purpose of the rule.

10 CFR 50.68 identifies criteria for ensuring special nuclear material (SNM) in the form of a fuel assembly remains sub-critical at nuclear power plants and that procedures exist to mitigate the event, if necessary. As discussed in Section 3 above and Section 3.12 of Enclosure 1 of this submittal, an evaluation of the limited number of fuel rods that will initially exceed the enrichment limit specified in 10

CFR 50.68(b) has been performed. The result of this assessment ensures the underlying purpose of the rule, subcriticality, will be maintained.

5.0 CONCLUSION

The 10 CFR 50.68(b) regulation is currently limited in applicability to the use of fuel rods with a maximum nominal U-235 enrichment limited to five (5.0) percent by weight. 10 CFR 50.68(b) does not apply to the proposed use of LTAs containing fuel rods with a maximum nominal U-235 enrichment of six percent by weight because the LTAs contain fuel rods with a maximum nominal U-235 enrichment greater than five (5.0) percent by weight. With the approval of this exemption request, this regulation will be applied to LTAs containing fuel rods with a maximum nominal U-235 enrichment of six percent by weight at the Vogtle Units 1 and 2 site.

In order to support the storage and use of four LTAs containing 4 fuel rods each with a maximum nominal U-235 enrichment of six percent by weight at VEGP, an exemption from the requirements of 10 CFR 50.68(b)(7) is requested. Pursuant to 10 CFR 50.12, the requested exemption is authorized by law, does not present undue risk to public health and safety, and is consistent with the common defense and security. Approval of this exemption request does not violate the underlying purpose of the rule. In addition, special circumstances exist to justify the approval of an exemption from the subject requirements.

6.0 REFERENCES

[1] Certificate of Compliance for Radioactive Material Packages No. 9380, April 7, 2022.

**Southern Nuclear Operating Company
Vogtle Electric Generating Plant – Units 1 and 2**

**License Amendment Request to Allow
Use of Lead Test Assemblies for Accident-Tolerant Fuel**

Enclosure 5

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 NL-22-0288 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: 6/17/2022

/Zachary Harper/
Signed electronically by
Zachary Harper