Constellation

4300 Winfield Road
Warrenville, IL 60555

630 657 2000 Office

WITHHOLD FROM DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

RS-22-097 10 CFR 50.90

August 31, 2022

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

Byron Station, Unit 2
Renewed Facility Operating License No. NPF-66
NRC Docket No. STN 50-455

Subject: License Amendment Request to Reinsert an Accident Tolerant Fuel Lead Test Assembly

Reference: Letter from J.S. Wiebe (U.S. NRC) to B.C. Hanson (Exelon Generation Company, LLC), *Byron Station, Unit No. 2 – Issuance of Amendment Regarding Use of Accident Tolerant Fuel Lead Test Assemblies (EPID L-2018-LLA-0064)*, dated April 3, 2019 (ADAMS Accession No. ML19038A017)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Constellation Energy Generation, LLC, (CEG) requests an amendment to Renewed Facility Operating License No. NPF-66 for Byron Station, Unit 2. This amendment request proposes to revise language in Technical Specification 2.1.1, "Reactor Core SLs," and 4.2.1, "Fuel Assemblies," to allow a previously irradiated Accident Tolerant Fuel (ATF) Lead Test Assembly (LTA) to be further irradiated during Byron Station Unit 2, Cycle 25.

CEG and Westinghouse Electric Company (Westinghouse) have embarked on a joint initiative to gather fuel performance data on Westinghouse accident tolerant fuel concepts in 2019 (See Reference). Byron Station plans to reinsert a previously irradiated LTA containing Westinghouse ADOPT™ with chromium-coated cladding test rods in Unit 2 during the Fall 2023 refueling outage. The subject LTA would remain in the Unit 2 core for one additional cycle, i.e., Cycle 25; and will then be discharged during the Spring 2025 refueling outage. This initiative will provide test data in support of developing a fuel solution that provides improvements in accident tolerance and fuel economics.

Attachments 6 and 7 contain Proprietary Information. Withhold from public disclosure under 10 CFR 2.390. When separated from Attachments 6 and 7 this document is decontrolled.

Attachments 1 and 6 to this letter provide information describing the proposed changes and a summary of the supporting analysis. Attachment 1 is a non-proprietary version of Attachment 6 where the proprietary information has been redacted. Attachment 3 provides the mark-up of the proposed Byron Station Technical Specifications changes. A clean copy of the proposed changes are provided in Attachment 4. Attachments 5 and 7 provide ORIGEN isotopic data related to the use of the LTA. Attachment 5 is a non-proprietary version of Attachment 7 where the proprietary information has been redacted.

Attachments 6 and 7 contain information proprietary to Westinghouse, and are supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit, found in Attachment 2, sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission (NRC) 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-22-040 and should be addressed to Camille T. Zozula, Manager, Regulatory Compliance & Corporate Licensing.

The proposed amendment has been reviewed by the Byron Station Plant Operations Review Committee in accordance with the requirements of the CEG Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), CEG is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State of Illinois official.

CEG requests approval of the proposed license amendment request within one year of this submittal date (i.e., by August 31, 2023), which supports loading the subject LTA in Byron Station, Unit 2 during the Fall 2023 refueling outage.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Rebecca L. Steinman at (630) 657-2831.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of August 2022.

Respectfully,

Lueshen, Kevin Digitally signed by Lueshen, Kevin
Date: 2022.08.31 07:05:26 -05'00'

Kevin Lueshen
Sr. Manager Licensing
Constellation Energy Generation, LLC

Attachments

1 Evaluation of the Proposed Change (**Non-Proprietary Version**)
2 Westinghouse Electric Company 10 CFR 2.390 Affidavit
3 Marked-up Byron Station Unit 2 Technical Specification Pages
4 Clean Byron Station Unit 2 Technical Specification Pages
5 ORIGEN Output Data **(Non-Proprietary Version)**
6 Evaluation of the Proposed Change (**Proprietary Version**)
7 ORIGEN Output Data **(Proprietary Version)**

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

RS-22-097 ATTACHMENT 1

BYRON STATION, UNIT 2

Docket No. STN 50-455

Renewed Facility Operating License No. NPF-66

Evaluation of Proposed Changes
(Non-Proprietary Version)

ATTACHMENT 1
Evaluation of Proposed Changes (Non-Proprietary Version)

Subject: License Amendment Request to Reinsert an Accident Tolerant Fuel Lead Test Assembly

ATTACHMENT 1
Evaluation of Proposed Changes (Non-Proprietary Version)

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Constellation Energy Generation, LLC, (CEG) requests an amendment to Renewed Facility Operating License No. NPF-66 for Byron Station, Unit 2. This amendment request proposes to revise language in Technical Specification (TS) 2.1.1, "Reactor Core SLs," and 4.2.1, "Fuel Assemblies," to allow a previously irradiated Accident Tolerant Fuel (ATF) Lead Test Assembly (LTA) to be further irradiated during Byron Station Unit 2, Cycle 25.

CEG and Westinghouse Electric Company (Westinghouse) have embarked on a joint initiative to gather fuel performance data on Westinghouse accident tolerant fuel concepts in 2019. Byron Station plans to reinsert a previously irradiated LTA containing Westinghouse Advanced Doped Pellet Technology (**ADOPT™**) with chromium-coated cladding test rods in Unit 2 during the Fall 2023 refueling outage. The subject LTA would be reinserted in the Unit 2 core for one additional cycle, i.e., Cycle 25; and will be discharged during the Spring 2025 refueling outage. This initiative will provide test data in support of developing a fuel solution that provides improvements in accident tolerance and fuel economics.

The currently licensed fuel design and reload analysis methods, including NRC-approved methods delineated in the COLR, do not fully accommodate the LTA/Lead Test Rod (LTR) design and materials; therefore, the Westinghouse analytical codes and methods will be supplemented, as necessary, using conservative assumptions and qualitative assessments based on test results, to confirm that all applicable limits associated with the LTA (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

2.0 DETAILED DESCRIPTION

The subject LTA is one of two assemblies, U75Y and U72Y, previously irradiated at Byron Station Unit 2. LTA U72Y is a Westinghouse VANTAGE+ Optimized Fuel Assembly design and when previously irradiated contained:

- Eight rods with standard uranium dioxide pellets and coated Optimized ZIRLO™ cladding
- Four rods with Westinghouse ADOPT™ uranium dioxide pellets and coated Optimized ZIRLO™ cladding
- All other rods have standard uranium dioxide pellets and standard Optimized ZIRLO™ cladding

LTA U72Y operated as expected in Cycle 22 and was reinserted in Cycle 23. It was discharged in the spring of 2022 and underwent poolside post-irradiation inspection / evaluation (PIE) a few months later. LTA U72Y will be reconstituted such that when reinserted in Cycle 25 it will contain:

- Seven stainless steel rods (including removal of one additional non-ATF rod for comparison)
- Four rods with standard uranium dioxide pellets and coated Optimized ZIRLO™ cladding
- Two rods with Westinghouse ADOPT™ uranium dioxide pellets and coated Optimized ZIRLO™ cladding
- All other rods have standard uranium dioxide pellets and standard Optimized ZIRLO™ cladding

Throughout the remainder of this document, the six chromium coated rods, with either ADOPT or standard pellets, will be referred to collectively as "ATF LTRs." The remaining fueled rods in the assembly will be referred to collectively as "standard LTRs," since these rods technically become LTRs when they exceed 62 GWd/MTU rod average exposure during Cycle 25. In cases where the delineation is not important, this document will refer to the entire test assembly instead of the test rods.

It is proposed to reinsert U72Y for Cycle 25 in the rodded center core assembly location to achieve burnups above the current accepted limits of use for the approved methods used to evaluate Byron Station Unit 2 fuel (62 GWd/MTU). This burnup will be applied to the entire assembly including both ATF and standard LTRs as well as the fuel assembly skeleton and associated assembly hardware. An assembly average of approximately **[ ] a,c** and a peak rod average of **[ ] a,c** burnup are projected.

The current safety limits defined in the Byron Station TS 2.0, with the exception of TS 2.1.1.3 for peak fuel centerline temperature, remain applicable to all the fuel assemblies, including the subject LTA, during Cycle 25. The peak fuel centerline temperature limit defined in TS 2.1.1.3 defines both a fresh fuel temperature limit and a rate of decrease limit. While the fresh fuel temperature limit is applicable to all the fuel assemblies, including the subject LTA, during Cycle 25; the rate of decrease limit that is currently specified is not applicable to LTA U72Y. The applicable rate of decrease limit for LTA U72Y is derived from PAD5 (Reference 11) which is used to model the projected high burnup. Therefore, this TS is modified to also include the associated rate of decrease limit which applies only to the LTA U72Y for Cycle 25. With this change, TS 2.1.1.3 states:

> In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU burnup for all assemblies except for U72Y for Cycle 25, which decreases by 9°F per 10,000 MWD/MTU burnup.

TS 4.2.1 currently contains a paragraph that allows for two LTAs containing up to a total of twenty test rods to be placed in the core during Cycles 22, 23, and 24. This statement requires the rods containing uranium silicide fuel pellets and standard UO_2 fuel pellets with coated Optimized ZIRLO™ cladding to be nonlimiting, and that the rods containing ADOPT™ pellets meet the fuel licensing limits under all conditions but are only required to be nonlimiting in steady-state conditions. This entire paragraph is being replaced with a new proposed paragraph that states:

> One LTA containing up to six Accident Tolerant Fuel (ATF) lead test rods may be placed in the Unit 2 reactor for evaluation. This LTA may be loaded in a core location that will

result in the LTA exceeding 62 GWd/MTU burnup at the end of Cycle 25. The LTA shall comply with the fuel limits specified in the COLR and Technical Specifications under all operational conditions.

Attachment 3 contains a marked-up version of the Byron Station, Unit 2 TS showing the proposed changes. Attachment 4 provides the revised (clean) TS pages.

3.0 TECHNICAL EVALUATION

The evaluation and description of the proposed Byron Station LTA is presented in the following sections. A representative loading pattern has been developed and sufficient technical analysis has been performed to evaluate the projected burnup for twice burned LTA U72Y and its placement in a rodded center core assembly location. The reload analysis will confirm these evaluations based on the final loading pattern.

3.1 Overview

The proposed license amendment requests approval for Byron Station, Unit 2 to reinsert LTA U72Y with six ATF LTRs containing either Westinghouse ADOPT™ fuel (uranium dioxide fuel pellets containing additions of chromium and aluminum oxides) and/or chromium coated fuel rod cladding.

The proposed LTA campaign supports the Westinghouse initiative to develop its ADOPT™ accident tolerant fuel. The Westinghouse ATF initiative is being performed pursuant to the U.S. Department of Energy program to develop light water reactor fuel types for the current fleet that have enhanced severe accident tolerance. Of particular importance are the areas of cladding strength and high temperature steam reaction kinetics, and fuel pellet thermal properties and fission product retention. Data acquired from LTA U72Y burnup will support improved fuel performance and fuel economics for ATF.

The evaluation below presents the technical justification and the regulatory basis supporting the conclusion that inserting the subject LTA in the Byron Unit 2 core for irradiation during Cycle 25 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.

The currently licensed NRC-approved methods for fuel design and reload analysis approved for use at Byron Station do not fully accommodate all design aspects and materials, and the anticipated LTA burnup will exceed the applicability limits of some of these codes and methods. Therefore, the Westinghouse analytical codes and methods were supplemented, as necessary, using sound engineering judgment and analytical codes and methods that reflect well-established engineering practices, and by conservatively addressing uncertainties in input parameters and models using the current state of knowledge and all available data to the extent practical. The analyses confirm that all applicable limits associated with the LTA (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as Shutdown Margin, transient analysis limits, and accident analysis limits) remain bounded by the current analysis of record. Furthermore, a requirement to demonstrate no cladding rupture is imposed to preclude concerns associated with Fuel Fragmentation, Relocation, and Dispersal (FFRD) under high burnup conditions.

3.2 Current Byron Station Unit 2 Core Configuration

A complete description of the Byron Station fuel system design basis can be found in Section 4.2, "Fuel System Design" of the Updated Final Safety Analysis Report (UFSAR) (Reference 1). Some key details are presented below.

The Byron Station Unit 2 core consists of 193 fuel assemblies. The core currently (i.e., Cycle 24) consists of three regions of Westinghouse VANTAGE+ Optimized Fuel Assemblies (OFAs) with Optimized ZIRLO™ cladding. Each fuel assembly typically consists of 264 fuel rods arranged in a 17x17 array, with a standard core inventory of 50,952 rods. These fuel assemblies are commonly referred to as "17 OFA."

The VANTAGE+ fuel rods consist of uranium dioxide ceramic pellets contained in Optimized ZIRLO™ cladding tubing, which is plugged and seal welded at the ends to encapsulate the fuel. The Optimized ZIRLO™ alloy is a zirconium alloy similar to Zircaloy-4, which has been specifically developed to enhance corrosion resistance. The VANTAGE+ fuel rods contain enriched uranium dioxide fuel pellets, and an integral fuel burnable absorber (IFBA) coating on some of the enriched fuel pellets.

Cycle 25 fuel assemblies will be of the same design (“17 OFA”) described above for the co-resident fuel assemblies. LTA U72Y containing six ATF LTRs will be loaded into Byron Station Unit 2 for irradiation during Cycle 25. There will be a combined total of 50,945 fuel rods in the core (accounting for the seven stainless steel rods in the LTA). The six ATF LTRs represent 0.012% of the core inventory; the two ADOPT™ rods represent 0.004% of the core inventory, and the four uranium dioxide fuel rods with coated Optimized ZIRLO™ cladding represent 0.008% of the core inventory. The 251 standard high burnup LTRs represent 0.49% of the core inventory.

3.3 Nuclear Safety and Design Considerations

The specific composition of LTA U72Y is detailed in Section 2.0. LTA U72Y has been evaluated using existing methods (to the extent practical) and sound engineering practices to demonstrate that use of the LTA in Cycle 25 poses no public health and safety concerns. It is recognized that the currently licensed fuel design and reload analysis methods for use at Byron Station do not fully accommodate the design and materials of U72Y, and the anticipated LTA burnup will exceed the applicability limits of some of these codes and methods. As a result, sound engineering judgment and analytical codes and methods that reflect well-established engineering practices are conservatively utilized to supplement existing codes and methods. Additionally, uncertainties in input parameters and models are conservatively addressed using the current state of knowledge and all available data to the extent practical. This modified set of analytical methods were then used to confirm that all applicable limits associated with LTA U72Y (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as Shutdown Margin, transient analysis limits and accident analysis limits) remain bounded by the current analysis of record.

LTA U72Y will be operated at a lower power than the lead assembly power; specifically, the relative power of LTA U72Y is projected to remain lower than the lead assembly under nominal conditions and will not lead the core during any postulated abnormal/accident condition. Since this LTA initiative involves a very limited number of ATF LTRs, it is expected (and will be confirmed prior to use as part of the reload analysis) that the LTA presents a negligible impact on reactor operation or nuclear safety. Impact on any aspect of reactor operation or safety will be negligible since the core design and reactor performance (during both normal and abnormal/accident operations) will be dominated by currently approved fuel types.

There are several design bases of the fuel system which are potentially impacted by the proposed LTA demonstration. The below design basis limits/criteria have been evaluated, and will be confirmed as necessary as part of the reload analysis, to demonstrate safe operation.

- The fuel rod cladding must exhibit satisfactory mechanical, material, and chemical properties, and must satisfy stress/strain and vibration/fatigue limits.
- The fuel pellet must exhibit satisfactory thermal physical and chemical properties, and dimensional, densification and swelling performance.
- The fuel rod must exhibit satisfactory pellet-clad mechanical interaction characteristics, pellet-clad gap and gas plenum dimensional stability, conformance to fuel temperature and internal gas pressure limits, heat transfer, fuel reliability, and overall dimensional stability.
- The fuel rod must be compatible with the overall fuel assembly design. The fuel rod must not compromise the performance or structural integrity of the fuel assembly, must not impair its ability to accommodate inserts such as rod cluster control assemblies (RCCAs), wet annular burnable absorbers (WABAs), secondary sources, and thimble plug assemblies and must not impair the performance of the reactivity control systems or the incore nuclear instrumentation.
- The test rods must not impair any aspect of neutronic behavior, including thermal margin, hot and cold reactivity, reactivity coefficients, reactor kinetics, and stability.
- The fuel rod and fuel assembly must be thermal-hydraulically compatible with the core and Reactor Coolant System (RCS) and must be compatible with all core and RCS materials and other plant equipment.
- The fuel assembly structural components (grids, top/bottom nozzle, thimble tubes, etc.) must satisfy all applicable criteria regarding stress/strain, vibration/fatigue, oxidation/hydriding, thermal hydraulic, and dimensional growth limits.

The modified Westinghouse analytical methods are capable of accurately modeling all aspects of neutronic behavior, including thermal margin, hot and cold reactivity, reactivity coefficients, reactor kinetics, and stability. All parameters associated with the fuel pellets and rods can be conservatively modeled to ensure that the margin of safety is not reduced. Additionally, given the very small number of ATF LTRs in the core, all parameters associated with core-wide neutronic design basis limits will be negligibly affected.

The thermal, physical, and chemical properties of the ADOPT™ fuel pellets are sufficiently understood to give a high level of confidence in the safety of the proposed activity. Sufficient design margin will be employed to ensure that pellet dimensional changes during operation will not pose a safety or operational concern. The ADOPT™ Fuel Description provided in Section 3.5 of the LAR application for Amendment No. 207 (Reference 3) remains accurate and complete.

Based on industry and testing experience to date, the performance of the test rods with respect to the shape, volume and function of the pellet-clad gap is well understood. The ability of the gap to accommodate fission product gases will not be affected, and there are no new pellet-clad interaction concerns introduced. Fuel temperature and pellet-clad heat transfer will be conservatively modeled and will not pose a safety or operational concern. There are no new fuel reliability concerns based on the results of the PIE performed after the end of Cycle 23. Additionally, it is projected that the fuel rods will perform well in all modes of operation and no adverse interactions with the current RCS chemistry regime are anticipated.

The ATF LTRs will not affect the performance of the host LTA and will be mechanically identical and compatible with the standard LTRs and co-resident fuel rods. The structural integrity of the assembly will be maintained and there will be no adverse effect on any piece of assembly hardware; therefore, the ability of the assembly to accommodate RCCAs and other inserts will not be affected. In particular, control rod motion will be unaffected during normal operation and transients, and the ability to control reactivity will be unaffected. The mechanical and nuclear function of the incore instrumentation will not be affected by the fuel rods, and there will also be no impact on the function or accuracy of the reactor protection system or the core monitoring system. The LTA will be thermal-hydraulically identical to the co-resident fuel, so there will be no impact to any aspect of core thermal-hydraulics or performance of the RCS. Accordingly, there will be no adverse impact affecting the interface with any plant equipment, including the reactor pressure vessel, fuel storage, fuel handling, and fuel inspection.

In summary, the reinserting of a single LTA containing a limited number of ATF LTRs in the Unit 2 core will not impact the public health and safety, and there will not be a significant impact on any aspect of normal plant operations, transient conditions, or accident analyses.

3.4 Technical Analysis

Mechanical Design Methodology

Westinghouse evaluated the following fuel assembly mechanical topics to confirm that LTA U72Y meets all current mechanical design criteria for reinsertion during Cycle 25 based on the PIE performed after the end of Cycle 23:

- Fuel Assembly Growth
- Hydraulic lift/ Holddown Force
- Fretting Wear
- Fuel assembly bow and RCCA Insertion
- Fuel Structural Component Integrity During Handling/Storage and Conditions I and II
- Fuel Structural Component Integrity During Conditions III and IV

The LTA U72Y fuel assembly length was measured after the end of cycle (EOC) 23. An estimate of the amount of additional growth projected to occur during Cycle 25 is based on existing fuel assembly growth methodology as a function of burnup. Based on the LTA U72Y fuel assembly length measurements at EOC 23 and the conservative evaluation, it has been confirmed that sufficient room is available between the core plates in order to accommodate the anticipated additional fuel assembly growth during Cycle 25.

The 17X17 OFA fuel design has historically been shown to exhibit satisfactory grid-to-rod fretting performance. An assessment has been completed using existing methodology to evaluate the risk of fretting when LTA U72Y is operated for a third cycle to high burnup. This assessment included factors such as fuel location, assembly power changes, residence time, higher burnup and best estimate flow rate through the assembly. Elevated-risk scenarios may include multiple cycles on the baffle, increased residence time, and a significant reduction in power of the assembly. None of these risk factors apply to the use of LTA U72Y in Cycle 25. The risk assessment concluded that since none of the elevated risk scenarios would be present during the third cycle of operation that the assembly was at low risk of developing a fretting related leaker.

The U72Y fuel assembly will be under a control rod assembly during Cycle 25. In order to demonstrate that the RCCA will be able to be fully inserted during Cycle 25, two sets of inspections were performed at the end of Cycle 23. The first inspection was a measurement of the drag load of an RCCA in LTA U72Y both inside the dashpot and above the dashpot. This drag load was compared to pre-defined limits intended to ensure that the RCCA can be inserted during the next cycle. The drag force limits for both the dashpot area and above the dashpot and corresponding drag force limit were met. The fuel assembly bow was also measured. Likewise, the assembly bow was also well below the pre-defined limit. Based on these measurements it is concluded that in the event of a SCRAM the RCCA drop time limits established by the overall reactor plant design basis will be met and the RCCA will be fully inserted.

The effect of the fuel assembly growth and corresponding increase in holddown force have been evaluated and found to be bounded by the increase in the material strength due to irradiation. The fuel assembly structural component corrosion has also been evaluated and found to be acceptable. Therefore, increased burnup does not affect the ability of the fuel assembly to be handled or to resist the loads during handling/storage or Conditions I and II.

The generic assessment using the standard methodologies demonstrated acceptable performance of the fuel at beginning of life (BOL) and end of life (EOL) conditions. Therefore, fuel rod and guide thimble stresses continue to meet allowable limits and RCCA insertability is maintained.

Core Physics

Westinghouse has employed a conservative nuclear design for the LTA. No adverse core physics impacts are predicted from the proposed activity.

In support of the nuclear design analysis, a representative loading pattern was developed with the LTA inserted into the center assembly location during its third cycle, resulting in high burnup values in this assembly. The maximum pin burnup anticipated in the LTA exceeded the current accepted limits of use for the approved methods used to evaluate Byron Station Unit 2 fuel (62 GWd/MTU) and is projected to reach approximately **[ ]** ^{a,c} at the end of Cycle 25. The LTA was shown to be non-limiting for power and peaking factors compared to the lead assemblies in the core. Safety parameters related to peaking factors and fuel melting in the LTA were analyzed explicitly. The core design models developed for this program were used both to support nuclear design evaluations and to provide power and burnup information to other disciplines.

The co-resident fuel was shown to behave similarly to a typical Byron Station design from a Core Physics perspective, and no other assemblies exceeded the current accepted limits of use for the approved methods used to evaluate Byron Station Unit 2 fuel (62 GWd/MTU). The representative loading pattern was designed to meet all the applicable design criteria. There were no changes to the standard overall nuclear design process in terms of incore fuel management, safety analyses, or operational data evaluation because of the proposed activity.

The current methods licensed for Byron Station Unit 2 (References 4 and 5) were used to model the core neutronics including the LTA. No changes to these methods were required to model the high burnup LTA. The ATF features are modeled explicitly. The ADOPT™ dopants and coating have a negligible neutronic impact, and only two fuel rods containing ADOPT™ fuel pellets are planned to be present in the LTA during Cycle 25. The reload analysis will confirm the nuclear design evaluations based on the final loading pattern

Loss-of-Coolant Accidents (LOCA)

An evaluation was performed to demonstrate that the LTA is non-limiting with respect to the existing Byron Station Unit 2 LOCA analyses using the methods currently utilized in Section 15.6.5 of UFSAR (Reference 1), which remain applicable for the evaluation of the LTA at higher burnup conditions. The presence of a small number of test rods will have an insignificant impact on the consequences of a postulated LOCA and analysis results for Peak Cladding Temperature (PCT), Maximum Local Oxidation (MLO), and Core Wide Oxidation (CWO). The LTA will have significant reductions of power and peaking factors relative to the core lead such that the LTA does not pose any additional FFRD risk. The assessment of the LTA design aspects and performance characteristics demonstrates that the 10 CFR 50.46 results will not be made more severe by the insertion of the LTA, and that the LTA remains bounded by the resident fuel comprising the existing LOCA analyses.

Not-LOCA Events

The not-Loss-of-Coolant-Accident (not-LOCA) events include the UFSAR (Reference 1) Chapter 15 non-LOCA analyses as well as the UFSAR (Reference 1) Chapter 6 analyses of steamline break (SLB) and LOCA mass and energy (M&E) releases for containment integrity. Two categories of not-LOCA events were considered for the high burnup (HBU) LTA and the ATF LTRs contained within the assembly: (1) those that are dependent on core-average effects and (2) those that are impacted by local effects in the fuel rods. Events dependent on core-average effects are negligibly impacted by the HBU LTA since the fuel rods contained within the LTA represent a small fraction of the total fuel rods contained in the core. Due to this, the change in core-average parameters used in the analyses, such as initial stored energy, core heat transfer characteristics, and decay heat, are insignificant. Events that are impacted by local effects in the fuel rods (e.g., hot rod, hot channel, or hot spot) could be affected more significantly and required more detailed consideration. Westinghouse completed an evaluation of the not-LOCA events using the not-LOCA methods currently utilized in the UFSAR (Reference 1), which remain applicable for the higher burnup conditions. Based on the placement of the LTA in a core location which was determined to be non-limiting for power and peaking factors, the conclusions documented in the applicable UFSAR (Reference 1) sections remain valid. Since the LTA will be operated at higher burnup values, it was demonstrated that the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the applicable limit value for the HBU LTA.

Thermal-Hydraulic

Westinghouse performed the thermal-hydraulic design evaluations for the HBU LTA using the existing methods applicable to Byron Station Unit 2 operating conditions and a representative reload design. All limits continue to be met and the LTA is bounded during operation by the existing plant safety analyses. The reload analysis will confirm these evaluations based on the final reload design. As described in Reference 1, the Departure from Nucleate Boiling (DNB) analysis of the VANTAGE+ fuel in Byron Station Unit 2 is based on the NRC-approved Revised Thermal Design Procedure (RTDP) methodology (Reference 6). The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. The primary DNB correlation used in the analysis of the VANTAGE+ fuel is the WRB-2 DNB correlation (Reference 7). The ABB-NV and WLOP DNB correlations (Reference 8) are used where the primary DNB correlation is not applicable. The ABB-NV correlation is applicable in the region below the first mixing vane grid. The WLOP DNB correlation is used for low pressure analyses. The VIPRE-W code (Reference 9) is used in the DNB analysis of Byron Station Unit 2.

The evaluations are intended to verify that the LTA is less limiting than the standard fuel rods with respect to thermal performance margin, and the LTA is hydraulically compatible with the resident fuel assemblies. The adequacy and conservative treatment of fuel thermal performance is reflected by the projected ample margin to the DNBR limit. Existing thermal-hydraulic design methods remain applicable and valid for the ATF and standard LTRs. Applicability of these codes will continue to be validated throughout the ATF development process by comparing the updated test data to the fuel design input. No adverse impact on thermal performance is anticipated, and all the test rods will be hydraulically compatible with the standard fuel rods.

For the Condition III and IV accident analyses, the current approved methods were used to confirm that the rods in the LTA will not be predicted to fail. Specifically, this was confirmed for the locked rotor and rod ejection accident analyses.

Rod bow evaluations were performed using the current licensed rod bow methodology (Reference 14) and associated gap closure correlations that were determined to be applicable to LTA U72Y. Results of the PIE rod bow measurements for LTA U72Y were within the expected range of the rod bow experience base with ample margin to existing gap closure correlation limit. Due to the relatively low power of the LTA, it will remain non-limiting at high burnup. The evaluation of the ATF LTR thermal performance was substantiated with comparative Critical Heat Flux (CHF) testing between coated and uncoated heater rods in the testing apparatus. It was verified that no CHF margin loss occurs due to the coating and no deterioration of surface heat transfer occurs. ATF LTR hydraulic compatibility was evaluated to verify that the coating surface roughness is similar to a standard fuel rod, and that no local hydraulic mismatch occurs due to a change in ATF LTR surface friction.

An assessment was performed to validate that the LTA thermal-hydraulic reload design evaluations remain bounded by the current analyses. A bounding current analysis will be confirmed, during the Cycle 25 reload, by placing the LTA in a core location to assure non-peak LTR power, verifying there is no change to the current DNB correlations and DNBR limits, and verifying that impacts to all other reload safety analysis and design inputs are negligible.

An evaluation was performed to confirm there are no adverse effects on the thermal-hydraulic design of the reload core due to the presence of the LTA. These evaluations included a comparison of surface roughness and friction between the test rods and standard rods, and a mechanical consistency and cooling check of key core components. Any impact on the cycle specific crud-induced power shift (CIPS) and crud-induced localized corrosion (CILC) analysis due to the LTA will also be assessed as part of the final reload process.

ATTACHMENT 1
Evaluation of Proposed Changes (Non-Proprietary Version)

Fuel Rod Design

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

- ADOPT™ Fuel Pellets
- Fuel Rod Chromium (Cr) Coating

The fuel performance features of ADOPT™ fuel are documented in Reference 10. 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 pellets.

Fuel rod Cr-coating provides improved corrosion resistance to the cladding; however, no corrosion benefits are taken for the fuel performance evaluations of the Byron Station Unit 2 Cycle 25 HBU LTA program. For the LTA program, the fuel rod Cr-coating is modeled with the same material properties and behaviors as the substrate material (Optimized ZIRLO™ cladding), with no credit taken for additional corrosion benefits. The chromium coating is modeled as uncoated Optimized ZIRLO™ rods 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 analyses.

Fuel performance calculations for the Byron Station Unit 2 Cycle 25 LTA considered the effects of the new materials using the latest set of fuel performance models, PAD5 (Reference 11). When necessary, changes were made to the PAD5 models and methods to analyze the new LTA fuel features. For ADOPT™ fuel, these changes were consistent with the as-submitted topical reports (References 10 and 11) 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 accepted limits of use for the approved methods used to evaluate Byron Station Unit 2 fuel (62 GWd/MTU). Although not approved beyond these limits in the NRC Safety Evaluation (SE) for Reference 11, the PAD5 fuel performance models were initially developed considering rod average burnups above **[ ]** a,c . Fuel performance data for rod average burnups beyond **[ ]** a,c were used in the calibration and validation of the models in the PAD5 code. PAD5 was used to perform the fuel rod design evaluations for any rod which exceeds 62 GWd/MTU burnup, up to and including **[ ]** a,c burnup. This LTA project will allow Westinghouse to further refine the PAD5 fuel performance models at high burnups.

The design limits will be confirmed using the latest fuel performance models from Reference 11, including NRC-approved input updates to model ADOPT fuel (Reference 10), as part of the standard reload analysis performed for Byron Station Unit 2 Cycle 25. All fuel performance criterion will be confirmed to be met for the Byron Station Unit 2 Cycle 25 LTA during projected operation of the fuel.

Fuel Handling and Storage

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. Adequacy of the tools, equipment, and procedures was demonstrated by prior handling of LTA U72Y. No new handling concerns are expected due to the additional irradiation cycle.

The reinserted LTA is not expected to have an impact on any aspect of the criticality analyses. This includes criticality safety for fuel storage and handling within the spent fuel pool. The previously performed evaluations for Region 1 Storage and Region 2 storage of the LTA remain valid for the reinserted LTA. Additional burnup experienced by the LTA will only further reduce the reactivity the LTA has during storage which is already acceptable for storage in Region 1 and Region 2. As a result, no additional impacts to fuel storage criticality result from reinsertion of the LTA.

Best Estimate Analyzer for Core Operations Nuclear (BEACON™) Core Monitoring System

Online core monitoring with the BEACON™ Core Monitoring System (i.e., the Power Distribution Monitoring System) will not be affected by the reinserted LTA, and the ability to accurately calculate the reactor 3-dimensional power shape will not be affected. Surveillances will be reliable and design basis peaking factor limits will be met at all times.

Seismic

The impact of the LTA on the seismic evaluation was previously evaluated to be negligible. The reinsertion of the LTA for an additional cycle has no impact on the prior seismic evaluation conclusions provided in Section 3.4 of the LAR application for Amendment No. 207 (Reference 3).

Alternate Source Term

Similar to the prior usage of LTA U72Y, the radiological source term will not be significantly affected by the inclusion of the ATF LTRs in the LTA or by the extension to higher burnup. Attachment 5 provides a tabular comparison of Byron's baseline inventory with an inventory including the HBU LTA. As described in Section 3.4 of the LAR application for Amendment No. 207 (Reference 3), the LTA is physically and chemically similar to the co-resident fuel so the mechanisms of release are not significantly different than those previously evaluated for the twice burned LTA. Since the core inventory, releases, failure mechanisms, and chemical structures between the baseline Alternative Source Term inventory and the LTA inventory are negligibly different or conservative in every instance, the radiological release limits for design basis accidents will not be challenged as a result of the additional irradiation cycle.

3.5 ADOPT™ Fuel Description

The use of ADOPT™ at Byron Station Unit 2 was found to be acceptable by the NRC staff per Amendment No. 207 (Reference 3). The ADOPT™ Fuel Description provided in Section 3.5 of the LAR application for Amendment No. 207 (Reference 3) remains accurate and complete.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires nuclear power reactors fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding to be provided with an emergency core cooling system with certain performance requirements. Although the Westinghouse ADOPT™ LTRs contain fuel and cladding material other than those defined in 10 CFR 50.46, the acceptance criteria specified in 10 CFR 50.46 will continue to be satisfied for the Byron Station Unit 2 core.

10 CFR Part 50, Appendix K, "ECCS Evaluation Models," Section I, "Required and Acceptable Features of the Evaluation Models," specifies the required attributes of the ECCS Evaluation Models. Paragraph A.1, "The Initial Stored Energy in the Fuel," states that, "the thermal conductivity of the UO_2 {uranium dioxide} shall be evaluated as a function of burn-up and temperature…" Paragraph I.A.5, "Metal-Water Reaction Rate," specifies that "the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation," where the Baker-Just equation applies specifically to the "zirconium-water" reaction. Based on the properties of the Westinghouse ADOPT™ LTRs, the results of the ECCS Evaluation Models for the resident fuel remain bounding when considering the impact of the ATF and standard LTRs. Moreover, the fuel rod Cr-coating is modeled with the same material properties and behaviors as the substrate material (Optimized ZIRLO™ cladding), with no credit taken for additional corrosion benefits.

"Regulatory Framework Applicability Assessment and Licensing Pathway Diagram for the Licensing of ATF-Concept, Higher Burnup, and Increased Enrichment Fuels" dated May 2022 provides the NRC staff's applicability determination of existing regulations and guidance for near-term ATF concept, higher burnup, and increased enrichment fuels. From Table 2-1, under "Burnup to 75 GWd/MTU" column, numerous guidance document regulations identify FFRD as the primary focus in relation to existing regulatory guidance. From Section 3, the LTA will have significant reductions of power and peaking factors relative to the core lead such that the LTA does not pose any additional FFRD risk. No cladding rupture has been demonstrated for the LTA to preclude concerns associated with FFRD under high burnup conditions.

4.2 Precedent

CEG, as well as other licensees, has previously conducted numerous LTA campaigns, including the prior irradiation of this LTA at Byron Station. Primary precedents applicable to the higher burnup aspects of this irradiation include a revision to the fuel rod average licensing basis burnup limit at Millstone Unit 3 for one LTA to a limit up to 71 GWd/MTU (Reference 12) as well as an exemption at V. C. Summer Unit 1 which allowed one LTA to continue to be irradiated up to a burnup of 75 GWd/MTU (Reference 13).

4.3 No Significant Hazards Consideration

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Constellation Energy Generation, LLC, (CEG) requests an amendment to Renewed Facility Operating License No. NPF-66 for Byron Station, Unit 2. This amendment request proposes to revise language in Technical Specifications 2.1.1, "Reactor Core SLs," and 4.2.1, "Fuel Assemblies," to allow a previously irradiated Accident Tolerant Fuel (ATF) Lead Test Assembly (LTA) to be further irradiated during Byron Station Unit 2, Cycle 25.

CEG and Westinghouse Electric Company (Westinghouse) have embarked on a joint initiative to gather fuel performance data on Westinghouse accident tolerant fuel concepts in 2019. Byron Station plans to reinsert a previously irradiated LTA containing Westinghouse ADOPT™ with chromium-coated cladding test rods in Unit 2 during the Fall 2023 refueling outage. The subject LTA would remain in the Unit 2 core for one additional cycle, i.e., Cycle 25; and will be discharged during the Spring 2025 refueling outage. This initiative will provide test data in support of developing a fuel solution that provides improvements in accident tolerance and fuel economics.

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 will be supplemented, as necessary, using conservative assumptions and qualitative assessments based on test results, to confirm that all applicable limits associated with the LTA (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) 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.

CEG has evaluated the proposed change for Byron Station, Unit 2 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?**

Response: No.

The proposed change involves the reinsertion of only a very small number of accident tolerant fuel (ATF) lead test rods (LTRs) which 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 projected burnup above 62 GWd/MTU associated with the LTA has a negligible impact on analytical results allowing the current analyses of record to remain bounding. The rodded center core assembly location for the LTA is non-limiting for power and peaking factors compared to the lead assemblies in the core. 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 assembly containing the ATF LTRs is the same design as the co-resident fuel in the core, with the exception of containing a limited number of ATF test rods or stainless steel rods in place of the standard fuel rods. The Byron Station Unit 2, Cycle 25 reload designs will meet all applicable design criteria. Evaluations of the LTA 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. Therefore, operation of the LTA will not significantly increase the predicted radiological consequences of accidents currently postulated in the Updated Final Safety Analysis Report.

Based on the above discussion, 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?**

Response: No.

The proposed change involves the use of a very small number of ATF LTRs in one LTA which is very similar in all aspects to the co-resident fuel. The proposed change does not alter 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 Byron Station Unit 2 reactor core 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 cycle in which the Westinghouse LTA will operate (i.e., Cycle 25) will demonstrate that the use of the LTA in the rodded center core location 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 the Westinghouse LTA does not involve any alteration to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors.

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?**

Response: No.

Operation of Byron Station Unit 2 with one Westinghouse LTA containing a limited number of ATF LTRs to achieve higher burnup levels 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 Byron Station Technical Specifications remain applicable to the subject LTA during Cycle 25. Westinghouse analytical codes and methods will be used, and supplemented as necessary using conservative assumptions, to confirm that all applicable limits associated with the LTA (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.

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.

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

Based on the above, CEG 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 reinsertion of one LTA containing a limited number of Westinghouse ADOPT™ with chromium-coated cladding accident tolerant fuel rods during Byron Station Unit 2, Cycle 25, will have a negligible impact on any aspect of reactor operations or reactor safety and 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.

5.0 ENVIRONMENTAL CONSIDERATION

CEG has evaluated the 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." CEG has determined that the proposed change to utilize one Lead Test Assembly (LTA) containing a limited number of Westinghouse ADOPT™ inside chromium-coated clad accident tolerant fuel rods during Byron Station Unit 2, Cycle 25, meets 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 the accident scenarios.

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 utilize one LTA containing a limited number of accident tolerant fuel rods during Byron Station Unit 2, Cycle 25, 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.

6.0 REFERENCES

1. Byron/Braidwood Stations, Updated Final Safety Analysis Report, Revision 18, dated December 2020

2. Byron Station Technical Specifications 2.1.1, Reactor Core SLs and 4.2.1, Fuel Assemblies

3. Byron Station, Unit No. 2 – Issuance of Amendment No. 207 Regarding Use of Accident Tolerant Fuel Lead Test Assemblies, April 2019 (ADAMS Accession No. ML19038A017)

4. WCAP-10965-P-A, ANC-A Westinghouse Advanced Nodal Computer Code, dated September 1986 (ADAMS Accession No. ML20211F881)

5. WCAP-11596-P-A, Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, dated June 1988 (ADAMS Accession No. ML20151M337)

6. WCAP-11397-P-A, Revised Thermal Design Procedure, dated April 1989 (ADAMS Accession No. ML20245A045)

7. WCAP-10444-P-A, Reference Core Report VANTAGE 5 Fuel Assembly, dated September 1985 (ADAMS Accession No. 8510090141)

8. WCAP-14565-P-A Addendum 2-P-A / WCAP-15306-NP-A Addendum 2-NP-A, Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications, dated April 2008 (ADAMS Accession No. ML081280711)

9. WCAP-14565-P-A (Proprietary) / WCAP-15306-NP-A (Non-Proprietary), VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, dated October 1999 (ADAMS Accession No. ML993160114)

10. WCAP-18482-P, Revision 0, Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel, dated May 2020 (ADAMS Accession No. ML20132A014)

11. WCAP-17642-P-A, Revision 1, Westinghouse Performance Analysis and Design Model (PAD5)," dated November 2017 (ADAMS Accession No. ML17334A826)

12. Millstone Power Station, Unit 3, License Amendment 228 regarding Lead Test Assembly, dated December 2005 (ADAMS Accession No. ML053200224)

13. V. C. Summer Nuclear Station – Exemption from the Requirements of 10 CFR Part 50, Sections 50.44 and Appendix K, dated March 2008 (ADAMS Accession No. ML080070466)

14. WCAP-8692, Revision 1, Fuel Rod Bow Evaluation, dated July 1979 (ADAMS Accession No. ML19208D506 and ML20079K315)

RS-22-097 ATTACHMENT 2

BYRON STATION, UNIT 2

Docket No. STN 50-455

Renewed Facility Operating License No. NPF-66

Westinghouse Electric Company 10 CFR 2.390 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 Attachments 6 and 7 to RS-22-097 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: 8/28/2022

Signed electronically by
Zachary Harper

RS-22-097 ATTACHMENT 3

BYRON STATION, UNIT 2

Docket No. STN 50-455

Renewed Facility Operating License No. NPF-66

Marked-up Byron Station Unit 2 Technical Specification Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell, ≥ 1.25 for the WRB-2 DNB correlation for a typical cell and ≥ 1.19 for the ABB-NV DNB correlation for a thimble cell and a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV DNB correlation and ≥ 1.18 for the WLOP DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU burnup.

for all assemblies except for U72Y for Cycle 25, which decreases by 9°F per 10,000 MWD/MTU burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

The site is located in Rockvale Township, approximately 3.73 mi (6 km) south-southwest of the city of Byron in northern Illinois.

4.1.2 Exclusion Area Boundary (EAB)

The EAB shall not be less than 1460 ft (445 meters) from the outer containment wall.

4.1.3 Low Population Zone (LPZ)

The LPZ shall be a 3.0 mi (4828 meter) radius measured from the midpoint between the two reactors.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions.

~~During Unit 2 Cycles 22, 23, and 24, two~~ One LTA~~s~~ containing up to ~~twenty total~~ six Accident Tolerant Fuel (ATF) lead test rods may be placed in the Unit 2 reactor for evaluation. This LTA may be loaded in a core location that will result in the LTA exceeding 62 GWd/MTU burnup at the end of Cycle 25. The LTA ~~The LTA rods containing uranium silicide fuel pellets and rods containing standard UO_2 fuel pellets with coated cladding shall be nonlimiting. The LTA rods containing ADOPT™ fuel pellets may be loaded in core regions which are nonlimiting under steady state reactor conditions and~~ shall comply with fuel limits specified in the COLR and Technical Specifications under all operational conditions.

RS-22-097 ATTACHMENT 4

BYRON STATION, UNIT 2

Docket No. STN 50-455

Renewed Facility Operating License No. NPF-66

Clean Byron Station Unit 2 Technical Specification Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell, ≥ 1.25 for the WRB-2 DNB correlation for a typical cell and ≥ 1.19 for the ABB-NV DNB correlation for a thimble cell and a typical cell.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.13 for the ABB-NV DNB correlation and ≥ 1.18 for the WLOP DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU burnup for all assemblies except for U72Y for Cycle 25, which decreases by 9°F per 10,000 MWD/MTU burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

The site is located in Rockvale Township, approximately 3.73 mi (6 km) south-southwest of the city of Byron in northern Illinois.

4.1.2 Exclusion Area Boundary (EAB)

The EAB shall not be less than 1460 ft (445 meters) from the outer containment wall.

4.1.3 Low Population Zone (LPZ)

The LPZ shall be a 3.0 mi (4828 meter) radius measured from the midpoint between the two reactors.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods or vacancies for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions.

One LTA containing up to six Accident Tolerant Fuel (ATF) lead test rods may be placed in the Unit 2 reactor for evaluation. This LTA may be loaded in a core location that will result in the LTA exceeding 62 GWd/MTU burnup at the end of Cycle 25. The LTA shall comply with fuel limits specified in the COLR and Technical Specifications under all operational conditions.

RS-22-097 ATTACHMENT 5

BYRON STATION, UNIT 2

Docket No. STN 50-455

Renewed Facility Operating License No. NPF-66

ORIGEN Output Data
(Non-Proprietary Version)

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

The Byron Station core inventory as affected by inclusion of U72Y lead test assembly (LTA) in the Unit 2 core is summarized in Table A-1. The isotopic inventory associated with just the LTA can be seen in Table A-2. The LTA assembly was modeled to a maximum burnup of **[ ]**[a,c] with an initial $^{235}_{92}\mathrm{U}$ enrichment of 4.401%, and a mass of 402.3 kg Uranium. The Table A-1 "Ratio" column demonstrates that the typical impact on any given nuclide is very small ($|\Delta| < 1\%$). The resulting core inventory as calculated by ORIGEN-ARP via the SCALE 6.1.2 software suite (Ref. A.3) is essentially unchanged from the non-LTA values.

The only nuclide with a change of greater than 1% is Curium-244 (Cm-244) ($^{244}_{96}\mathrm{Cm}$) which experienced an increase of 3.1%. The buildup of Cm-244 is highly dependent on exposure with "production of curium-244 is found to increase with about the fourth power of burnup" (Ref. A.1). Since this is the third irradiation of the LTA, the observed increase in predicted Cm-244 aligns with expectations. Given the non-LOCA total release fraction of the Lanthanide group is zero and the LOCA release fraction is just 0.0002 (Ref. A.2) a change of 3.1% does not cause any significant increase in the calculated dose consequences.

References

A.1 SAND-95-1990C, Ewing, R I., *Burnup verification measurements at U.S. Nuclear Facilities using the Fork system*, dated September 1, 1995 (Available from https://www.osti.gov/biblio/110688-burnup-verification-measurements-nuclear-facilities-using-fork-system)

A.2 Regulatory Guide 1.183, Revision 0, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, dated July 2000 (ADAMS Accession No. ML003716792)

A.3 *Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2011. (Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785)

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

Table A-1 Byron Core Inventory with High Burnup LTA Included

Group	Group Name	Nuclide	AST Inventory[1] (Ci/Core)	Reduced AST (AST x 192/193)[2] (Ci/Core)	Adjusted AST (Reduced AST + LTA[3]) (Ci/Core)	Ratio
1	Noble Gases	Kr-85	1.08E+06	1.07E+06	1.08E+06	1.002
		Kr-85m	3.10E+07	3.08E+07	3.09E+07	0.997
		Kr-87	6.23E+07	6.20E+07	6.21E+07	0.997
		Kr-88	8.43E+07	8.39E+07	8.41E+07	0.997
		Xe-133	1.96E+08	1.95E+08	1.96E+08	0.999
		Xe-135	5.47E+07	5.45E+07	5.47E+07	0.999
2	Halogens	I-131	9.63E+07	9.58E+07	9.62E+07	0.999
		I-132	1.41E+08	1.40E+08	1.41E+08	0.999
		I-133	2.01E+08	2.00E+08	2.01E+08	0.999
		I-134	2.30E+08	2.28E+08	2.29E+08	0.999
		I-135	1.90E+08	1.89E+08	1.90E+08	0.999
3	Alkali Metals	Cs-134	1.65E+07	1.64E+07	1.66E+07	1.008
		Cs-136	4.91E+06	4.89E+06	4.92E+06	1.002
		Cs-137	1.13E+07	1.13E+07	1.14E+07	1.003
		Rb-86	1.95E+05	1.94E+05	1.95E+05	1.004
4	Tellurium Group	Sb-127	8.76E+06	8.71E+06	8.76E+06	1.000
		Sb-129	2.72E+07	2.71E+07	2.72E+07	0.999
		Te-127	8.59E+06	8.54E+06	8.58E+06	0.999
		Te-127m	1.43E+06	1.42E+06	1.42E+06	0.997
		Te-129	2.55E+07	2.54E+07	2.55E+07	1.000
		Te-129m	4.88E+06	4.86E+06	4.88E+06	0.999
		Te-131m	1.86E+07	1.85E+07	1.86E+07	0.999
		Te-132	1.38E+08	1.37E+08	1.37E+08	0.999
5	Strontium	Sr-89	1.04E+08	1.04E+08	1.04E+08	0.998
		Sr-90	8.35E+06	8.31E+06	8.37E+06	1.002
		Sr-91	1.44E+08	1.43E+08	1.44E+08	0.997
		Sr-92	1.50E+08	1.49E+08	1.50E+08	0.997
6	Barium	Ba-139	1.84E+08	1.83E+08	1.84E+08	0.998
		Ba-140	1.79E+08	1.78E+08	1.79E+08	0.998

[1] AST Inventory represents the current Byron Alternate Source Term analysis of record inventory as calculated by ORIGEN-ARP.
[2] Reduces the core by one assembly.
[3] The LTA's inventory from Table A-2 is added in to make the Adjusted AST inventory which is the final model of the core including both non-LTA and LTA fuel. The "Maximum" column from Table A-2 is used for conservatism.

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

Table A-1 Byron Core Inventory with High Burnup LTA Included

Group	Group Name	Nuclide	AST Inventory[1] (Ci/Core)	Reduced AST (AST x 192/193)[2] (Ci/Core)	Adjusted AST (Reduced AST + LTA[3]) (Ci/Core)	Ratio
7	Noble Metals	Mo-99	1.83E+08	1.82E+08	1.83E+08	0.999
		Rh-105	9.65E+07	9.60E+07	9.65E+07	1.000
		Ru-103	1.52E+08	1.51E+08	1.52E+08	1.000
		Ru-105	1.06E+08	1.05E+08	1.06E+08	1.000
		Ru-106	5.68E+07	5.65E+07	5.70E+07	1.003
		Tc-99m	1.62E+08	1.61E+08	1.61E+08	0.999
8	Cerium	Ce-141	1.62E+08	1.61E+08	1.62E+08	0.999
		Ce-143	1.63E+08	1.62E+08	1.63E+08	0.998
		Ce-144	1.24E+08	1.24E+08	1.24E+08	1.000
		Np-239	1.85E+09	1.84E+09	1.85E+09	1.000
		Pu-238	3.18E+05	3.16E+05	3.21E+05	1.009
		Pu-239	2.91E+04	2.89E+04	2.91E+04	1.000
		Pu-240	4.14E+04	4.12E+04	4.15E+04	1.003
		Pu-241	1.27E+07	1.26E+07	1.27E+07	1.002
9	Lanthanides	Am-241	1.30E+04	1.30E+04	1.31E+04	1.004
		Cm-242	3.67E+06	3.65E+06	3.70E+06	1.009
		Cm-244	4.14E+05	4.12E+05	4.27E+05	1.031
		La-140	1.83E+08	1.82E+08	1.83E+08	0.999
		La-141	1.68E+08	1.67E+08	1.67E+08	0.998
		La-142	1.64E+08	1.63E+08	1.64E+08	0.998
		Nb-95	1.65E+08	1.64E+08	1.65E+08	0.999
		Nd-147	6.59E+07	6.56E+07	6.58E+07	0.999
		Pr-143	1.59E+08	1.58E+08	1.58E+08	0.998
		Y-90	8.70E+06	8.66E+06	8.72E+06	1.002
		Y-91	1.29E+08	1.28E+08	1.29E+08	0.998
		Y-92	1.52E+08	1.51E+08	1.51E+08	0.997
		Y-93	1.65E+08	1.65E+08	1.65E+08	0.998
		Zr-95	1.63E+08	1.62E+08	1.63E+08	0.999
		Zr-97	1.73E+08	1.72E+08	1.73E+08	0.998

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

Table A-2 Isotopic Inventory of the High Burnup LTA

Group	Group Name	Nuclide	Total at 100 EFPD (Ci/ Assembly)	Total at EOC (Ci/ Assembly)	Maximum (Ci/ Assembly)	48-hour Decay (Ci/ Assembly)
1	Noble Gases	Kr-83m	3.90E+04	3.62E+04	3.90E+04	1.84E-01
		Kr-85	6.67E+03	7.47E+03	7.47E+03	7.47E+03
		Kr-85m	7.85E+04	7.04E+04	7.85E+04	4.24E+01
		Kr-87	1.49E+05	1.31E+05	1.49E+05	5.76E-07
		Kr-88	1.94E+05	1.69E+05	1.94E+05	1.38E+00
		Xe-133	7.56E+05	7.56E+05	7.56E+05	6.71E+05
		Xe-135	2.02E+05	1.90E+05	2.02E+05	4.29E+04
		Xe-131m	5.07E+03	5.24E+03	5.24E+03	5.11E+03
		Xe-133m	2.46E+04	2.48E+04	2.48E+04	1.80E+04
		Xe-135m	1.75E+05	1.78E+05	1.78E+05	8.05E+02
		Xe-138	6.34E+05	6.22E+05	6.34E+05	0.00E+00
2	Halogens	Br-84	6.34E+04	5.74E+04	6.34E+04	3.22E-23
		Br-85	7.75E+04	6.92E+04	7.75E+04	0.00E+00
		I-129	2.03E-02	2.63E-02	2.63E-02	2.63E-02
		I-131	3.92E+05	3.95E+05	3.95E+05	3.42E+05
		I-132	5.68E+05	5.71E+05	5.71E+05	3.70E+05
		I-133	7.74E+05	7.68E+05	7.74E+05	1.59E+05
		I-134	8.55E+05	8.45E+05	8.55E+05	1.01E-10
		I-135	7.44E+05	7.41E+05	7.44E+05	4.69E+03
3	Alkali Metals	Cs-134	1.71E+05	2.25E+05	2.25E+05	2.24E+05
		Cs-136	2.65E+04	3.67E+04	3.67E+04	3.30E+04
		Cs-137	7.61E+04	9.44E+04	9.44E+04	9.44E+04
		Cs-138	7.00E+05	6.88E+05	7.00E+05	1.29E-20
		Rb-86	1.37E+03	1.76E+03	1.76E+03	1.63E+03
		Rb-88	1.99E+05	1.75E+05	1.99E+05	1.55E+00

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

Table A-2 Isotopic Inventory of the High Burnup LTA

Group	Group Name	Nuclide	Total at 100 EFPD (Ci/ Assembly)	Total at EOC (Ci/ Assembly)	Maximum (Ci/ Assembly)	48-hour Decay (Ci/ Assembly)
4	Tellurium Group	Sb-127	4.18E+04	4.37E+04	4.37E+04	3.09E+04
		Sb-129	1.22E+05	1.27E+05	1.27E+05	6.67E+01
		Te-127	3.78E+04	3.92E+04	3.92E+04	3.15E+04
		Te-127m	3.11E+03	2.96E+03	3.11E+03	3.00E+03
		Te-129	1.17E+05	1.21E+05	1.21E+05	1.27E+04
		Te-129m	2.12E+04	2.07E+04	2.12E+04	2.00E+04
		Te-131m	7.95E+04	8.20E+04	8.20E+04	3.02E+04
		Te-132	5.52E+05	5.53E+05	5.53E+05	3.59E+05
5	Strontium	Sr-89	3.08E+05	2.34E+05	3.08E+05	2.28E+05
		Sr-90	4.95E+04	5.67E+04	5.67E+04	5.67E+04
		Sr-91	3.56E+05	3.18E+05	3.56E+05	1.01E+04
		Sr-92	4.01E+05	3.66E+05	4.01E+05	1.71E+00
6	Barium	Ba-137m	7.24E+04	8.98E+04	8.98E+04	8.94E+04
		Ba-139	6.66E+05	6.56E+05	6.66E+05	2.70E-05
		Ba-140	6.41E+05	6.26E+05	6.41E+05	5.61E+05
7	Noble Metals	Mo-99	7.02E+05	6.99E+05	7.02E+05	4.22E+05
		Rh-105	4.91E+05	5.31E+05	5.31E+05	2.41E+05
		Ru-103	7.55E+05	7.33E+05	7.55E+05	7.08E+05
		Ru-105	5.34E+05	5.81E+05	5.81E+05	3.33E+02
		Ru-106	4.55E+05	4.80E+05	4.80E+05	4.33E+05
		Tc-99m	6.22E+05	6.21E+05	6.22E+05	4.08E+05
8	Cerium	Ce-141	6.35E+05	5.86E+05	6.35E+05	5.64E+05
		Ce-143	5.40E+05	5.22E+05	5.40E+05	1.92E+05
		Ce-144	6.78E+05	5.39E+05	6.78E+05	5.37E+05
		Np-239	9.10E+06	9.72E+06	9.72E+06	5.44E+06
		Pu-238	2.96E+03	4.60E+03	4.60E+03	4.61E+03
		Pu-239	1.61E+02	1.60E+02	1.61E+02	1.61E+02
		Pu-240	2.86E+02	3.20E+02	3.20E+02	3.20E+02
		Pu-241	8.70E+04	9.39E+04	9.39E+04	9.39E+04

ATTACHMENT 5
ORIGEN Output Data (Non-Proprietary Version)

Table A-2 Isotopic Inventory of the High Burnup LTA

Group	Group Name	Nuclide	Total at 100 EFPD (Ci/ Assembly)	Total at EOC (Ci/ Assembly)	Maximum (Ci/ Assembly)	48-hour Decay (Ci/ Assembly)
9	Lanthanides	Am-241	8.52E+01	1.14E+02	1.14E+02	1.14E+02
		Cm-242	3.78E+04	5.30E+04	5.30E+04	5.28E+04
		Cm-244	6.59E+03	1.48E+04	1.48E+04	1.48E+04
		La-140	6.85E+05	6.77E+05	6.85E+05	6.27E+05
		La-141	5.99E+05	5.87E+05	5.99E+05	1.31E+02
		La-142	5.68E+05	5.53E+05	5.68E+05	1.90E-04
		Nb-95	7.54E+05	5.50E+05	7.54E+05	5.50E+05
		Nd-147	2.44E+05	2.42E+05	2.44E+05	2.14E+05
		Pr-143	5.26E+05	5.07E+05	5.26E+05	4.89E+05
		Y-90	5.19E+04	5.97E+04	5.97E+04	5.85E+04
		Y-91	4.29E+05	3.26E+05	4.29E+05	3.20E+05
		Y-92	4.06E+05	3.70E+05	4.06E+05	1.24E+02
		Y-93	4.79E+05	4.47E+05	4.79E+05	1.72E+04
		Zr-95	6.79E+05	5.48E+05	6.79E+05	5.37E+05
		Zr-97	6.14E+05	6.02E+05	6.14E+05	8.26E+04