UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 20, 2023

Mr. David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS NOS. 233 AND 233 RE: TECHNICAL SPECIFICATIONS 2.1.1 AND 4.2.1 TO ALLOW A PREVIOUSLY IRRADIATED ACCIDENT TOLERANT FUEL LEAD TEST ASSEMBLY TO BE FURTHER IRRADIATED IN UNIT NO. 2 (EPID L-2022-LLA-0131)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 233 to Renewed Facility Operating License No. NPF-37 and Amendment No. 233 to Renewed Facility Operating License No. NPF-66 for the Byron Station (Byron), Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated August 31, 2022 (Agencywide Documents Access Management System (ADAMS) Accession No. ML22243A094), as supplemented by letter dated February 27, 2023 (ML23058A147).

The amendment revises the Technical Specifications (TS) 2.1.1, “Reactor Core SLs [Safety Limits],” and 4.2.1, “Fuel Assemblies”. The proposed changes will allow a previously irradiated Accident Tolerant Fuel (ATF) Lead Test Assembly (LTA) to be further irradiated during Byron, Unit 2, Cycle 25. No technical changes are made to the Byron, Unit 1, Operating License. The Byron, Unit 1, Amendment No. is administratively incremented because the TSs are common to both units.

Enclosure 3 to this letter contains proprietary information. When separated from Enclosure 3, this document is DECONTROLLED.

A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454 and STN 50-455

Enclosures:
1. Amendment No. 233 to NPF-37
2. Amendment No. 233 to NPF-66
3. Proprietary Safety Evaluation
4. Nonproprietary Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
Renewed License No. NPF-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Constellation Energy Generation, LLC (the licensee) dated August 31, 2022, as supplemented by letter dated February 27, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-37 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed
Facility Operating License and
Technical Specifications

Date of Issuance: July 20, 2023

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
Renewed License No. NPF-66

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Constellation Energy Generation, LLC (the licensee) dated August 31, 2022, as supplemented by letter dated February 27, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-66 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 233, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: July 20, 2023

ATTACHMENT TO LICENSE AMENDMENT NOS. 233 AND 233

RENEWED FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454 AND STN 50-455

Replace the following pages of the Renewed Facility Operating Licenses and Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating Licenses

REMOVE	INSERT
License NPF-37	License NPF-37
-3-	-3-
License NPF-66	License NPF-66
-3-	-3-

Technical Specifications

REMOVE	INSERT
2.0 – 1	2.0 – 1
4.0 – 1	4.0 – 1

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 233, and the Environmental Protection Plan contained in Appendix B, both of which were attached to Renewed License No. NPF-37, dated November 19, 2015, are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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.

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-37 AND

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-66

CONSTELLATION ENERGY GENERATION, LLC

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454 AND STN 50-455

(NON-PROPRIETARY)

Proprietary information pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* has been redacted from this document.

Redacted Proprietary information is identified by empty double brackets.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NON-PROPRIETARY SAFETY EVALUATION BY

THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE

NO. NPF-37 AND AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING

LICENSE NO. NPF-66

CONSTELLATION ENERGY GENERATION, LLC

BYRON STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454 AND STN 50-455

1.0 INTRODUCTION

By letter dated August 31, 2022, Constellation Energy Generation, LLC, (CEG, the licensee) submitted a license amendment request (LAR) to request changes to Technical Specifications (TSs) 2.1.1, "Reactor Core SLs [Safety Limits]," and 4.2.1, "Fuel Assemblies." The proposed changes will allow a previously irradiated accident tolerant fuel (ATF) lead test assembly (LTA) to be further irradiated during Byron Station (Byron), Unit No. 2, Cycle 25 (Reference 1). The licensee plans to reinsert a previously irradiated LTA, containing test rods with Westinghouse ADOPT™ fuel pellets and chromium-coated cladding, in Byron, Unit No. 2, during the Fall 2023 refueling outage (RFO). 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 RFO. No technical changes are made to the Byron, Unit 1, Operating License. The Byron, Unit 1, Amendment No. is administratively incremented because the TSs are common to both units. The licensee supplemented the letter dated August 31, 2022, by letter dated February 27, 2023 (Reference 2), which provided responses to NRC staff requests for additional information (RAIs).

The February 27, 2023, supplement contained clarifying information and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's initial proposed finding of no significant hazards consideration as published in the *Federal Register* on December 6, 2022 (87 FR 74668).

2.0 REGULATORY EVALUATION

The purpose of Title 10 of the Code of Federal Regulations (10 CFR) 50.46 and appendix K to 10 CFR part 50 is to establish acceptance criteria for the emergency core cooling systems (ECCS) performance. The regulations in 10 CFR 50.46 and appendix K contain acceptance criteria for the ECCS for reactors fueled with zircaloy or ZIRLO™ fuel rod cladding material.

However, the licensee has previously obtained an exemption for Byron Units 1 and 2, allowing application of the acceptance criteria of 10 CFR 50.46 and 10 CFR part 50, appendix K, to fuel assembly designs using Optimized ZIRLO™ fuel rod cladding material (Reference 5).

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 ECCS with certain performance requirements.

10 CFR 50.46(b)(4), "Coolable geometry," states that "Calculated changes in core geometry shall be such that the core remains amenable to cooling." For the present LAR, this requirement is particularly relevant to ensuring that fuel fragmentation, relocation, and dispersal (FFRD) phenomena would not result in unacceptable geometric changes to the high burnup LTA.

10 CFR part 50, appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC 35, "Emergency core cooling," states, in part, that "A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."

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 I.A.1 of appendix K to 10 CFR part 50, "The Initial Stored Energy in the Fuel," states, in part, that, "the thermal conductivity of the UO2 [uranium dioxide] shall be evaluated as a function of burnup and temperature…" Similarly, 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.

10 CFR 50.67, "Accident source term [AST]," establishes radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. Specifically, the regulation states, in part, that:

> (2) The NRC may issue the amendment [revising the accident source term for applicable licensees] only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) An individual located at any point on the boundary of the exclusion area [EAB] for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone [LPZ], who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room [CR] under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) [TEDE] for the duration of the accident. (footnote omitted).

10 CFR part 50, appendix A, General Design Criteria for Nuclear Power Plants, Criterion 19--Control room, states that:

> A control room shall 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 [LOCAs]. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

10 CFR 50.36(c)(1) requires technical specifications to include safety limits. Safety limits are "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity."

10 CFR 50.36(c)(4) requires that technical specifications include design features. Design features are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered under the safety limits, limiting conditions for operation or surveillance requirement categories.

Regulatory Guide (RG) 1.183 (Reference 11) provides the methodology for analyzing the radiological consequences of several design-basis accidents (DBAs) to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of AST submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

On September 8, 2006, the NRC issued Amendment No. 147 to Facility Operating License No. NPF-37 and Amendment No. 147 to Facility Operating License No. NPF-66 for the Byron, Unit Nos. 1 and 2, respectively (Reference 12). The amendments fully implemented an AST, pursuant to 10 CFR 50.67 "Accident source."

3.0 TECHNICAL EVALUATION

3.1 Description

The Byron, Unit No. 2, core consists of 193 fuel assemblies, with each assembly typically consisting of 264 fuel rods arranged in a 17x17 array. The current cycle, Cycle 24, consists of three regions of Westinghouse VANTAGE+ Optimized Fuel Assemblies (OFAs) with Optimized ZIRLO™ cladding. The VANTAGE+ fuel rods consist of enriched UO2 ceramic pellets contained in Optimized ZIRLO™ cladding tubing. The tubing is plugged, and seal welded at the ends to encapsulate the fuel. There is an integral fuel burnable absorber (IFBA) coating on some of the enriched fuel pellets. With the exception of certain design features of the LTA that are described further below, Cycle 25 fuel assemblies are planned to be of the same design as described for the co-resident fuel assemblies. A complete description of the Byron fuel system design basis can be found in section 4.2, "Fuel System Design" of the Updated Final Safety Analysis Report (UFSAR) (Reference 3).

The licensee plans to reinsert a previously irradiated LTA (i.e., U72Y) containing six ATF lead test rods (LTRs) for further irradiation in Cycle 25. The subject LTA is a Westinghouse VANTAGE+ Optimized fuel assembly design. The assembly has been previously irradiated in Cycles 22 and 23 at Byron, Unit No. 2. It was discharged in Spring of 2022 following Cycle 23, and underwent poolside post-irradiation evaluation (PIE) a few months later.

During the previous irradiation in Cycles 22 and 23 at Byron, Unit No. 2, the LTA U72Y contained:

- eight rods with standard UO2 pellets and coated Optimized ZIRLO™ cladding,
- four rods with Westinghouse ADOPT™ UO2 pellets and coated Optimized ZIRLO™ cladding, and
- the remaining rods with standard UO2 pellets and standard Optimized ZIRLO™ cladding.

The licensee plans to reconstitute the subject LTA for Cycle 25 to contain:

- four rods with standard UO2 pellets and coated Optimized ZIRLO™ cladding,
- two rods with Westinghouse ADOPT™ UO2 pellets and coated Optimized ZIRLO™ cladding,
- seven stainless-steel rods, which replace four rods with standard pellets and coated cladding, two rods with ADOPT™ pellets and coated cladding, and one additional non-ATF rod, and
- the remaining rods with standard UO2 pellets and standard Optimized ZIRLO™ cladding.

The licensee is proposing to reinsert the reconstituted LTA U72Y for Cycle 25 in the rodded center core location. The LTA will achieve burnups above the currently accepted limit of 62 gigawatt-days per metric ton of uranium (GWd/MTU) for the approved fuel analysis methods for Byron, Unit No. 2. This burnup will be applied to the entire LTA, including both ATF and standard rods, as well as the other fuel assembly components. The projected assembly average burnup for the subject LTA is approximately **[[** **]]** with a projected peak rod average of **[[** **]]**.

3.2 Proposed Technical Specification Changes

The licensee proposed revising Section 2.1.1, "Reactor Core SLs" of their TS. Specifically, Section 2.1.1.3 would be revised to address the rate of decrease limit for the high burnup U72Y fuel assembly during cycle 25. All other safety limits would be unchanged by this amendment. Section 2.1.1.3 would be revised to read as follows (Note: the new text is shown in italics):

> 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 2.1.1.3 defines the peak fuel centerline temperature limit for both fresh fuel and a rate of decrease to this limit as fuel burnup increases. The rate of decrease limit currently specified in TS 2.1.1.3 is not applicable to LTA U72Y for Cycle 25. Therefore, the licensee proposed a change in the rate of decrease limit as applicable to LTA U72Y to model the projected high burnup during Cycle 25. The applicable rate of decrease limit for the high burnup LTA U72Y in Cycle 25 is derived using the NRC-approved PAD5 code. This rate of decrease limit is only applicable to the subject LTA for Cycle 25 reinsertion.

The licensee also proposed a revision to TS 4.2.1 "Fuel Assemblies." The revision would delete the description of the placement of two LTAs in the core during Cycles 22, 23, and 24 and replace it with a description of one LTA being re-inserted in the core for Cycle 25. The description states that the LTA will exceed the 62GWd/MTU burnup limit. Specifically, the proposed revision would replace the last paragraph in section 4.2.1 with the following:

> 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.

The current TS 4.2.1 at Byron allows for two LTAs containing up to a total of 20 test rods to be placed in the core during Cycles 22, 23, and 24. The TS places a requirement that the rods containing uranium silicide fuel pellets and standard UO2 fuel pellets with coated Optimized ZIRLO™ cladding must be non-limiting under steady-state conditions. It further states that the rods containing ADOPT™ pellets are required meet the fuel licensing limits under all conditions. The final LTA guidance letter issued by the NRC in 2019 (Reference 4), states that to meet the TS provision of the non-limiting core region, the licensee should perform an evaluation to demonstrate that the location of LTA in the core along with the respective operating parameters ensures that any new design features of the LTA maintain more thermal and mechanical margin to the design, performance, and safety limits (SLs) relative to the co-resident fuel during normal operation, anticipated operational occurrences, and postulated accidents. The NRC staff determined that the licensee has satisfied the intent of the letter by submitting its request for prior NRC staff review via a LAR.

3.3 Technical Evaluation

3.3.1 Fuel Design Description

The planned LTA will contain test rods with Westinghouse ADOPT™ fuel pellets and chromium-coated cladding. The use of ADOPT™ pellets at Byron, Unit No. 2, was previously approved by the NRC staff in License Amendment No. 207 (Reference 6). With the exception of the reconstitution-related modifications to the LTA that have been described above (i.e., addition of seven inert, stainless-steel rods), the NRC staff finds the fuel description provided in Amendment No. 207 applicable to the current LAR. As discussed in WCAP-13060P-A (Reference 7), the limited use of inert replacement rods is consistent with Westinghouse's approved fuel assembly reconstitution methodology.

3.3.2 Mechanical Design

The fuel mechanical design evaluation for the LTA U72Y was performed by the licensee to confirm all current mechanical design criteria remain applicable and meet the specified limits for reinsertion during Cycle 25. The evaluation was done based on the PIE performed at the end of Cycle 23. The design criteria evaluated in the LAR are discussed below:

Fuel Assembly Growth: The licensee measured the fuel assembly length at the end of Cycle 23. Projected estimated growth during Cycle 25 was performed using the existing fuel assembly growth methodology. The licensee's evaluation showed that, based on the LTA measurements at end of cycle (EOC) 23 and the projected Cycle 25 growth, there is sufficient room available between the core plates to accommodate the fuel assembly growth during reinsertion in Cycle 25.

Based on its review, the NRC staff finds the licensee's evaluation acceptable because it was performed using the existing methodology and has conservative growth projections for Cycle 25.

Hydraulic Lift/ Holddown Force: The licensee performed an evaluation of the potential increase in holddown force due to fuel assembly growth and states that it is bounded by an increase in material strength due to irradiation. The licensee evaluated corrosion of the fuel assembly structural components and found it to be acceptable.

Limits on hydraulic lift loads are placed such that upward hydraulic forces do not exceed the combined weight of the assembly and the downward force of the holddown springs. While fuel assembly growth with burnup increases the holddown force, the holddown springs also exhibit relaxation with increasing burnup. These are competing effects. The Byron UFSAR (Reference 3), section 4.1, references WCAP-12610-P-A, appendix B (Reference 8), which describes in part the design methodology for the VANTAGE+ fuel assembly. Appendix B of WCAP-12610-P-A states that holddown springs are designed to keep fuel assemblies in contact with the lower core plate during all Conditions I and II events. The discussion therein further finds that fuel assembly growth will continue to compensate for additional spring relaxation as a function of fluence. Section 4.4.2.6.2 of the Byron UFSAR further states that hydraulic loads at normal operating conditions are calculated using the mechanical design flow for the minimum core bypass flow. Similarly, the hydraulic loads at cold plant startup conditions are adjusted to account for the density difference of the coolant. For the pump overspeed transient, which could create flow rates 20 percent greater than the design flow, the hydraulic loads are evaluated to be approximately twice the fuel assembly weight.

The licensee has proposed no changes to design basis criteria for hydraulic lift and holddown for the LTA. Based on the evaluations performed by the licensee, which considered the competing effects of high burnup on the holddown force from the fuel assembly growth versus the holddown spring relaxation, and the conservatisms described in the Byron, Unit No. 2, UFSAR for the hydraulic load calculations, the NRC staff finds the holddown forces will not be challenged due increased burnup during Cycle 25. The NRC staff finds reinsertion of LTA U72Y in Cycle 25 will not affect the ability of the fuel assembly to be handled or resist loads during fuel handling or Conditions I and II events because of the small number of LTRs inserted compared to the core inventory and the minimal changes in assembly design characteristics of the LTA.

<u>Fretting Wear</u>: An assessment using the existing methodology to evaluate the risk of fretting for LTA U72Y operating for a third cycle at high burnup was performed by the licensee. The assessment included factors such as the location of the fuel assembly, power fluctuations, residence time, burnup, and the flow rate through the assembly. The licensee further evaluated conditions known to result in an elevated risk of fretting, including loading an assembly for multiple cycles adjacent to the core baffle, a significant reduction in assembly power and increased residence time. The licensee asserted that none of the risk factors apply to LTA U72Y in Cycle 25 and concluded that LTA U72Y was at low risk of developing fretting wear during operation in Cycle 25.

Based on the NRC staff's review of the licensee's evaluations and historically satisfactory performance of 17x17 OFA fuel for grid-to-rod fretting, the NRC staff finds that LTA U72Y will not exhibit elevated fretting risk during Cycle 25. The NRC staff also notes that while the LTA will have significant reductions of power and peaking factors relative to the core lead, this power reduction would be bounded by the reduction associated with the relocation of interior fuel assemblies to the core periphery.

<u>Fuel Assembly Bow and RCCA Insertion</u>: The licensee performed two sets of inspections at the end of Cycle 23 to demonstrate that a rod cluster control assembly (RCCA) will be able to be fully inserted into LTA U72Y during Cycle 25. The inspections compared drag loads to pre-defined limits determined to ensure that an RCCA can be inserted and further measured for the fuel assembly bow. Based on these inspections, the licensee concluded that in the event of a scram, existing RCCA Drop Time limits (e.g., TS SR 3.1.4.3) will be met and the RCCA will be capable of full insertion.

Based on its review, the NRC staff finds the licensee's evaluation for the fuel assembly bow and RCCA insertion in the event of a scram to be acceptable because licensee inspections indicate that TS SR 3.1.4.3 will be met. In addition the RCCA drop time will be verified to meet TS SR 3.1.4.3 prior to taking the reactor critical following placement of LTA U72Y into the reactor as required by TS SR 3.1.4.3.

<u>Fuel Structural Component Integrity</u>: Evaluations of the structural integrity of fuel assembly components during handling, storage, and Conditions I, II, III and IV, events were performed by the licensee. Based on these evaluations of various fuel assembly mechanical components, the

licensee concluded that 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. For the Conditions III and IV accident analyses, the licensee used currently approved methods to confirm that the rods in the LTA would not be predicted to fail for postulated events.

Based on a review of the results of the licensee-performed evaluations included in the LAR, the NRC staff finds that fuel structural component integrity will not be challenged during handling, storage, and postulated Conditions I, II, III and IV, events.

The NRC staff also evaluated the design criteria evaluations performed in the LAR. Based on its review, the NRC staff concludes that the mechanical design of the high burnup LTA U72Y is acceptable for the proposed reinsertion in Cycle 25 up to a projected assembly average burnup of **[[ ]]** and a projected peak pin average burnup of **[[ ]]** for Cycle 25.

3.3.3 Core Physics

The licensee developed a representative loading pattern with LTA U72Y inserted into the center assembly location during its third cycle, Cycle 25. As discussed above, the maximum pin burnup anticipated in the LTA exceeds the currently accepted limits from the approved methods used to evaluate Byron, Unit No. 2, fuel (62 GWd/MTU) and is projected to reach approximately **[[ ]]** at the end of Cycle 25.

The NRC staff has generally applied two criteria (Reference 4) for LTA programs, i.e., the number of LTAs should be limited and the core locations of LTAs should be non-limiting. While the subject LTA is being inserted into the center assembly location, the LTA will be non-limiting for power and peaking factors compared to the lead assemblies in the core. The parameters related to peaking factors and fuel melting of the LTA are analyzed in the LAR. The licensee's core physics evaluations show that the LTA behaves similar to a typical Byron reload fuel assembly. The representative core loading pattern was designed to meet all the applicable design criteria. The licensee stated that there were no changes to the overall nuclear design process including fuel management, safety analyses or operational data evaluation. The licensee stated that ATF features of the LTA were explicitly modeled using current methods licensed for Byron, Unit No. 2. The licensee stated that the two fuel rods containing ADOPT™ fuel pellets with dopants (i.e., chromia (Cr_2O_3)) and alumina (Al_2O_3) and a coating (i.e., ZrB_2 absorber) that are planned to be present in the LTA during Cycle 25 have a negligible neutronic impact.

Based on the licensee's conservative nuclear design employed for the LTA, the use of existing methodologies licensed for Byron, Unit No. 2, and the similarity of the LTA design to typical reload assemblies, the NRC staff finds that there are no adverse core physics impacts from reinsertion of LTA U72Y in Cycle 25. The staff notes that the licensee will perform reload analysis to confirm the nuclear design evaluations based on the final core loading pattern.

3.3.4 Loss-of-Coolant Accidents (LOCAs)

The licensee performed an evaluation in the LAR to demonstrate the LTA U72Y is non-limiting with respect to the existing Byron, Unit No. 2, LOCA analysis. While the subject LTA is being inserted into the center assembly location, the LTA is non-limiting for power and peaking factors compared to the lead assemblies in the core. The licensee performed its evaluation using methods described in section 15.6.5 of the UFSAR (Reference 3). The analysis concluded that the presence of a small number of test rods will not have a significant impact of consequences of a postulated LOCA. Specifically, the analysis results for the relevant quantitative figures of merit from 10 CFR 50.46(b), peak cladding temperature (PCT), maximum local oxidation and core wide oxidation, did not show any significant impact.

The NRC staff agrees with the licensee's finding that the LOCA analysis results will not be made more severe by the insertion of the LTA due to: (a) the power and peaking factors of the LTA being non-limiting compared to the core lead, and (b) the presence of a small number of test rods compared to the total number of fuel rods present in the core. Further, the design characteristics of the LTA are shown to be similar to the rest of the fuel assemblies. Based on its review of the licensee's evaluations, the NRC staff finds that the LTA remains bounded by the existing LOCA analysis for the resident fuel.

LOCA FFRD Concerns Evaluation

During a LOCA transient, high burnup fuel has the potential to fragment, and these fragments could axially relocate within the fuel rod. Should a rupture of the fuel rod cladding occur, a further potential exists that these fragments could be expelled from the fuel rod burst opening into the coolant. These phenomena are collectively referred to as FFRD. Any fuel fragments expelled from the rod would be expected to be small in size, and it may be possible that some fraction of these fragments could transport beyond the reactor vessel, reach the location of the break in the reactor coolant system pressure boundary, and enter the containment. The presence of fuel outside an array-like core geometry has not been evaluated and could challenge the licensees' existing evaluations for core coolability, containment cooling, equipment qualification, radiological dose and other evaluations considered in the plant safety analysis.

The NRC staff in Research Information Letter (RIL) 2021-13 (Reference 9), evaluated existing experimental research and defined a conservative, empirical threshold for FFRD-related phenomena. The RIL states that data from existing research has shown that fine fragmentation is limited to fuel above 55 GWd/MTU burnup. The RIL further states that fuel fragmentation and the potential for fuel dispersal increases as burnup increases. Therefore, with respect to the proposed LAR, the licensee needs to demonstrate that the potential for the high burnup LTA to experience FFRD to an extent that would result in noncompliance with existing regulatory requirements may reasonably be precluded.

In section 3.4 of attachment 6 to the LAR, the licensee states that the LTA will have a significant reduction of power and peaking factors relative to the core lead assembly. The licensee further states in section 4.1 of attachment 6 to the LAR that no cladding rupture has been demonstrated for the LTA, which would preclude concerns associated with fuel dispersal under high burnup conditions. The NRC staff agrees with the licensee that one approach for addressing concerns related to FFRD is to demonstrate that fragmentation-susceptible rods will not rupture. However, the NRC staff notes that the licensee's assessment that the LTA would not experience rupture was based upon a comparison of the limiting rods in the LTA to fuel rods in an analyzed assembly that was similar but not identical to the LTA. The key differences observed by the NRC staff were associated with the linear heat generation rate, rod internal pressure (RIP), and fuel stored energy. As identified in RAI 2, these differences between the characteristics of the LTA and the analyzed fuel rod the licensee used to assess the LTA could influence the margin to rupture.

In response to this RAI, by letter dated February 27, 2023, the licensee provided additional discussion on the FFRD concerns and the evaluations performed to address them (Reference 2). The licensee stated that the LTA will be reinserted in the center core assembly, underneath a guide tube (GT) structure. The licensee further stated that **[[** **]]**. The licensee stated that the LTA's average linear heat rate (ALHR) varies between **[[** **]]**, with the **[[** **]].** The LTA's peak linear heat rate (PLHR) varies between **[[** **]]** and reaches the **[[** **]]**. The burnup of the lead rod in the LTA is expected to range from **[[** **]].** In the licensee presented calculations **[[** **]]**.

Since the RIP and fuel rod stored energy are of particular interest when demonstrating margin to rupture, the licensee performed a comparison at **[[** **]]**. The licensee used a **[[** **]]** to be conservative. Although the LTA rods contain both UO2 and ADOPT® pellets, **[[** **]]**. The comparisons presented by the licensee showed that the RIP for the LTA fuel and the analysis of record (AOR) fuel are similar up to the 62 GWd/MTU burnup limit. However, the predicted RIP range at end of life for the LTA is **[[** **]]** than the AOR fuel.

An increase in RIP increases the potential for rod burst. To account for this detrimental effect, the licensee performed a margin-to-rupture analysis by **[[** **]]**. The licensee compared the fuel average temperature from the AOR fuel performance to those for the LTA rods at the PLHR values mentioned above. The comparison showed that **[[** **]]**. The licensee did not include the beneficial effect of fuel stored energy in calculating the margin to rupture, as an added conservatism.

The licensee performed an assessment of the margin to rupture for the limiting LTA rod for the LOCA event to demonstrate that fuel dispersal will not occur. To calculate the margin, **[[** **]]**.

The licensee also performed the same calculations based on a **[[**

]]. These calculations showed that margin to rupture is maintained, despite not considering the beneficial effect of a lower fuel stored energy in the LTA. The licensee further observed **[[**

]].

The NRC staff finds the licensee's calculations performed for the projected RIPs and the margin to rupture cases to be acceptable. The RIP projected range for the **[[ ]]** AOR calculations bounds the LTA values, and the margin to rupture is maintained in the calculations performed by the licensee using adequate conservatisms. In particular, margin was shown for the limiting **[[ ]]**, and substantial margin is indicated for conditions more representative of 95/95 tolerance limit values that the NRC has found acceptable for demonstrating compliance with 10 CFR 50.46 for analyses of emergency core cooling system performance that use realistic methods with an explicit accounting for uncertainty.

Based on the calculations presented for the RIP and the margin to rupture by the licensee, including the factors that could affect the propensity for rupture, the lower power and peaking factors of the LTA being reinserted, and the relatively small fraction of the LTA fuel rods compared to the total core inventory, the NRC staff finds that the licensee's proposed reinsertion of the LTA does not present undue risk of fuel dispersal. The NRC staff, however, notes that the determination of the acceptability of the high burnup LTA reinsert with respect to FFRD concerns is based on the Byron, Unit No. 2 specific calculations and justifications only and may not be applicable to conditions beyond those considered in this LAR.

3.3.5 Non-LOCA and Chapter 6 Events

The licensee identified non-LOCA events included in Chapter 15 of the UFSAR (Reference 3) and the analyses of steam line break (SLB) and LOCA mass and energy (M&E) releases for the containment integrity included in the Chapter 6 of the UFSAR as 'not-LOCA' events in the LAR. Two categories of events were considered; those that are dependent on core-average effects, and those that are only impacted by the local effects in the fuel rods.

The licensee concluded that the events dependent on core-average effects are negligibly impacted by a high burnup LTA insertion, given the small fraction of the LTA rods inserted compared to the total number of typical reload fuel rods in the core. The licensee's evaluation concluded that the changes in core-average parameters related to initial stored energy, core heat transfer characteristics, and decay heat are insignificant.

There are a total of 251 standard high burnup LTRs, which represent only 0.49 percent of the total core inventory of 50,945 fuel rods. Further, the six ATF LTRs, the two ADOPT™ rods and the four UO2 fuel rods with coated Optimized ZIRLO™ cladding, make up an even smaller fraction of the total core inventory, as presented in section 3.2 of the LAR. Given the small fraction of the LTA rods inserted compared to the core inventory, the NRC staff finds the licensee's conclusion that events dependent upon core-average properties will be negligibly impacted by the insertion of a high burnup LTA to be acceptable. Therefore, the SLB and LOCA M&E releases, which depend on the core-average parameters, will also be negligibly impacted. As such containment pressures and temperatures, which in turn depend on the SLB and LOCA M&E releases, will also be negligibly impacted, and thus, the containment integrity will be negligibly impacted.

For the events that are impacted by local effects in the fuel rods (e.g., limited by a hot rod, hot channel, or hot spot), the licensee performed an evaluation of the non-LOCA events using the methods currently utilized in the UFSAR (Reference 3). The evaluation demonstrated that the minimum departure from nucleate boiling ratio (DNBR) for the LTA being operated at the higher burnup values will remain above the applicable limit (i.e., DNB [departure from nucleate boiling] will be avoided), and that the core location of the LTA is non-limiting with respect to the assembly power and peaking factors.

The NRC staff agrees with the licensee's evaluation for non-LOCA events limited by local effects since the LTA will be inserted in a location non-limiting for power and peaking, and the licensee has demonstrated that the DNBR limit will not be exceeded. Based on its review, the NRC staff finds that the methods used are applicable for high burnup conditions and the conclusions documented in the UFSAR remain valid for the LTA reinsertion.

Based on the analyses performed by the licensee, the NRC staff concludes that the high burnup LTA U72Y will meet the acceptance criteria for the non-LOCA events in the UFSAR chapter 15 as well as the chapter 6 analyses of SLB and LOCA M&E releases for assuring containment integrity.

The NRC staff finds the non-LOCA analysis and chapter 6 analyses of the SLB and LOCA M&E release to be acceptable for the planned reinsertion of the LTA to an approximate assembly average burnup of **[[ ]]** with projected peak pin average of **[[ ]]** for Cycle 25.

3.3.6 Thermal-Hydraulic Analysis

The licensee performed thermal-hydraulic design evaluations for the high burnup LTA using existing methods applicable to Byron, Unit No. 2, operating conditions. A representative core design was used in the analysis. The LAR describes that the DNB analysis of the VANTAGE+ fuel in Byron, Unit No. 2, is largely based on the NRC-approved revised thermal design procedure (RTDP) methodology. The standard thermal design procedure (STDP) is used for the analyses where the RTDP is not applicable. The primary DNB correlation used in the analysis of the VANTAGE+ fuel is the WRB-2 DNB correlation. The licensee applies other correlations where the primary DNB correlation is not applicable. In particular, the ABB-NV correlation is used for the axial region of the core below the first mixing vane grid, and the WLOP DNB correlation is used for the analysis of events involving low pressures below the WRB-2 correlation's qualification range. The licensee's evaluations were performed using NRC-approved VIPRE-W subchannel code.

The evaluations performed by the licensee demonstrate that the LTA is less limiting than the standard fuel rods with respect to thermal performance margin. They also demonstrate that the LTA is hydraulically compatible with the resident fuel assemblies. Through the evaluations, the licensee concluded that existing thermal-hydraulic design methods remain applicable to the

LTA, including the ATF LTRs. Rod bow evaluations were performed using the currently licensed methodology and associated gap closure correlations. The licensee determined these methods are applicable to LTA U72Y. The results from the licensee's evaluations are within the expected range of the rod bow experience base with margin to the existing gap closure correlation limit.

The licensee performed an evaluation of the ATF LTRs' thermal performance with critical heat flux (CHF) testing between coated and uncoated heater rods in the testing apparatus. The evaluation showed no CHF margin loss or deterioration of surface heat transfer occurs due to the coating. The licensee evaluated ATF LTR hydraulic compatibility 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. Evaluations were performed to validate that the LTA thermal-hydraulic reload design evaluations remain bounded by the existing analyses and there are not adverse effects on thermal-hydraulic design of the reload core due to the presence of the LTA.

The licensee plans to confirm the current analyses during the Cycle 25 reload by verifying that there is no change to the current DNB correlations and DNBR limits. The licensee also plans to verify that impacts to all other reload safety analysis and design inputs are negligible during the Cycle 25 reload. The impact on the cycle specific crud-induced power shift (CIPS) and crud-induced localized corrosion (CILC) analysis due to the LTA insertion is planned to be addressed as part of the final reload process.

For Conditions III and IV events, the licensee indicated that the accident analysis performed for the rods in the LTA in Cycle 25 using currently approved methods predicts no failures. In response to the NRC staff RAIs, by letter dated February 27, 2023, the licensee provided additional discussion on the methodology and results from evaluations performed for the locked rotor and the rod ejection accidents (Reference 2).

Locked Rotor Analysis

For the existing locked rotor analysis, the licensee used the **[[** **]]**. The fuel temperatures are expected to increase at high burnups due to the thermal conductivity degradation (TCD) making it non-conservative from the DNBR perspective**.** The licensee performed an evaluation of the LTA for the locked rotor event **[[** **]]**. Based on the calculations, the licensee concluded that there is **[[** **]]**. The licensee concluded that no rods in the LTA will experience DNB since the **[[** **]]**.

Rod Ejection Analysis

For the rod ejection analysis, the licensee performed an evaluation based on the existing method in the Byron UFSAR. **[[**

]]. The licensee performed rod ejection calculations for the Cycle 25 LTA using the **[[ ]]**. The analysis was performed for **[[ ]]**. From the evaluations performed, the licensee concluded that there is sufficient **[[ ]]**, and thus the DNB design basis is met for the LTA reinserted in Cycle 25.

Based on the analysis presented in the LAR and subsequent response to the RAIs, the NRC staff find the conclusions drawn from the licensee's thermal-hydraulic analyses to be acceptable. The licensee used approved methodologies and codes where applicable, and demonstrated margins to fuel failure criteria exist. Therefore, the staff finds that reasonable assurance exists that the high burnup fuel rods in the LTA would not experience fail. The licensee performed the thermal-hydraulic analyses using a representative core reload design and plans to confirm these evaluations based on the final reload design for Cycle 25. The NRC staff finds the licensee's plan to verify the representative reload design with the final Cycle 25 reload design to be acceptable.

3.3.7 Fuel Rod Design

The ADOPT™ fuel is a modified UO2 pellet doped with small amounts of chromia (Cr_2O_3) and alumina (Al_2O_3). The licensee stated that these additives lead to greater densification and diffusion during sintering. This results in a higher density and an enlarged grain size as compared to undoped UO2 fuel pellets. The licensee performed fuel performance calculations for the ADOPT™ fuel in LTA U72Y for Byron, Unit No. 2, Cycle 25, considering the effects of the new materials using the NRC-approved PAD5 code, with appropriate changes to the PAD5 models to accommodate the ADOPT™ fuel. The licensee indicates that the changes to the PAD5 models were made in a manner consistent with the as-submitted topical report WCAP-18482-P for the ADOPT™ fuel (Reference 10) and the subsequent NRC RAIs. The licensee did not take any corrosion-resistance credit for the Cr-coating despite the expectation of improved corrosion resistance.

Some rods in the LTA assembly are intended to exceed the 62 GWd/MTU burnup limit from the approved methods for Byron, Unit No. 2. The licensee used the fuel performance data for rod average burnups beyond **[[ ]]** in the calibration and validation of the models in the PAD5 code. The code was used to perform the fuel rod design evaluations for rods exceeding 62 GWd/MTU burnup limit, up to **[[ ]]** burnup. As part of the standard reload analysis performed for Byron, Unit No. 2, Cycle 25, the licensee plans to confirm the design criteria and limits using the latest fuel performance models, which include the ADOPT™ fuel input updates approved by the NRC.

The NRC staff finds use of the PAD5 code for performing fuel rod design evaluations beyond the approved limits of 62 GWd/MTU for the LTA rods to be acceptable given:

a) the small fraction of the LTRs being modeled compared to the entire core inventory,

b) the use of the NRC-approved PAD5 code beyond its approved value being limited only to the LTA assembly and the modifications to PAD5 being consistent with the

ADOPT topical report (WCAP-18482-P) and the subsequent information provided by the licensee in response to NRC staff RAIs as part of this LAR review, and

c) the licensee-demonstrated compatibility and margins for the design characteristics, mechanical, thermal-hydraulic and accident analyses, in addition to the fuel rod design evaluations performed using the PAD5 code.

3.3.8 Fuel Handling and Storage

The licensee demonstrated adequacy of the fuel handling tools, equipment, and procedures through prior handling of LTA U72Y during previous cycles. There are no changes anticipated in the interface of LTA U72Y with other plant related equipment or any changes the handling tools or procedures. Based on the licensee's confirmation of no changes to the LTA interface or the procedures and tools for handling the fuel, the NRC staff finds the fuel handling during the additional cycle of irradiation to be acceptable.

The licensee previously performed evaluations for storing LTA U72Y in Region 1 and Region 2 of the spent fuel pool. These analyses will remain valid for the reinserted LTA for criticality because additional burnup of LTA U72Y is expected to further reduce its reactivity. Based on its review, the NRC staff agrees with the licensee's finding of no additional impacts to fuel storage criticality resulting from reinsertion of LTA U72Y and, therefore, finds it acceptable.

3.3.9 Core Monitoring System

The Best Estimate Analyzer for Core Operations Nuclear (BEACON™) system is used for the core monitoring. The NRC staff agrees with the licensee's statement that the core monitoring system will not be affected by LTA reinsertion since the monitoring system was used in previous cycles with the LTA, and there is no impact in the ability of the monitoring system to accurately calculate 3-dimensional power shapes in the reactor core.

3.3.10 Seismic

The impact of the LTA on the seismic evaluation was previously evaluated as part of license Amendment No. 207 (Reference 6) and was found to be negligible. Based on its review of the LAR, the NRC staff finds that reinsertion of LTA U72Y for an additional cycle will have no impact on the previous seismic evaluation given the small number of LTRs inserted compared to the core inventory and the minimal changes in assembly design characteristics of the LTA. Therefore, the NRC staff finds the seismic evaluation to be acceptable.

3.3.11 Alternate Source Term (AST)

Radiological DBAs

Reference 6 approved the regulatory and technical analyses, inputs, and assumptions related to the radiological consequences of DBAs performed by the licensee in support of initial insertion of the LTAs. The NRC staff reviewed the assumptions, inputs, and methods, used by the licensee to assess the impacts of the LAR.

The previously approved amendment which supported the initial insertion of the LTAs added descriptive text to TS 4.2.1 that authorized use of two LTAs containing a limited number of ATF LTRs during Byron, Unit No. 2, Cycles 22, 23 and 24; to be discharged during the Fall 2023 RFO. These LTAs were placed in non-limiting core locations. The two LTAs contained a combined total of 20 LTRs. For comparison, there are a combined total of 50,952 fuel rods in the Byron, Unit No. 2, core. The combined total of 20 LTRs represented 0.039 percent of the core inventory. Of these 20 LTRs, the four fuel rods containing uranium silicide pellets represent 0.008 percent of the core inventory and the pellets occupy approximately one foot of each rod; the 12 fuel rods with standard uranium dioxide fuel pellets and coated Optimized ZIRLO cladding represent 0.024 percent of the core inventory; and the four fuel rods containing ADOPT fuel pellets represent 0.008 percent of the core inventory.

The current LAR is requesting reinsertion of one of the original LTAs, identified as “LTA U72Y.” This LTA operated as expected in Cycle 22 and was reinserted in Cycle 23. It was discharged in Spring of 2022 and will be reconstituted prior to reinsertion in Cycle 25. The singular LTA, when reconstituted will contain: four rods with standard UO2 pellets and coated Optimized ZIRLO cladding; two rods with Westinghouse ADOPT UO2 pellets and coated Optimized ZIRLO cladding; seven stainless steel rods, and all other rods in the LTA will have standard UO2 pellets and standard Optimized ZIRLO cladding. As noted by the licensee, the percent representation of the LTA rods to be reinserted during Cycle 25 will be as follows:

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 percent of the core inventory; the two ADOPT™ rods represent 0.004 percent of the core inventory, and the four UO2 fuel rods with coated Optimized ZIRLOTM cladding represent 0.008 percent of the core inventory. The 251 standard high burnup LTRs represent 0.49 percent of the core inventory.

The proposed LAR notes that the current licensing basis radiological source term will not be significantly affected by the re-insertion of the ATF LTRs in the LTA or by the extension to higher burnup. The change in the proposed Byron core inventory with high burn up with the LTAs included and current licensing basis is negligible. The nuclide with the largest expected change of greater than 1 percent is curium-244 (Cm-244) which experienced an increase of 3.1 percent. The non-LOCA fraction of fission product inventory in the gap of the Lanthanide group is zero and the LOCA core inventory fraction released into containment is just 0.0002 percent as provided in RG-1.183. An increase of 3.1 percent in Cm-244 would negligibly contribute to control room or offsite doses. The remainder of the source term nuclides increase by less than 1 percent, or decrease, and would not contribute significantly to control room or offsite doses.

The NRC staff reviewed the change in the Byron core inventory with the high burnup LTA and found that the proposed amendment would not cause any significant increase in calculated radiological consequence analysis and that the requirements of 10 CFR 50.67 and 10 CFR 50, appendix A, GDC 19, are met and that the guidance in RG 1.183 is met.

3.3.12 Technical Conclusion

The NRC staff reviewed the licensee’s methodology and evaluations provided in the LAR and the supplementary submittal in response to the staff’s RAIs for revisions to TSs 2.1.1, “Reactor Core SLs,” and 4.2.1, “Fuel Assemblies,” to allow a previously irradiated ATF LTA containing test rods with Westinghouse ADOPT™ fuel pellets and chromium-coated cladding to be further irradiated during Byron, Unit No. 2, Cycle 25.

Based on the evaluations presented above, the NRC staff finds that the licensee's analysis methods and evaluations adequately demonstrate that reinserting high burnup LTA U72Y for further irradiation during Cycle 25 will have no adverse impact on any aspect of reactor operations or reactor safety. The NRC staff finds that the analyses performed using a representative reload design remain bounded by the current AOR. The licensee plans to verify the analyses performed using a representative reload design with the final reload design for Cycle 25. Therefore, the NRC staff finds the licensee plan to be acceptable.

The acceptability of different aspects of the nuclear analysis, including the fuel description and rod design, aspects of the mechanical design, core physics, LOCA and non-LOCA analyses, thermal-hydraulic design with considerations for locked rotor and rod ejection events, fuel handling and storage considerations, core monitoring, seismic considerations, and the AST considerations are discussed individually above. Further, based on the licensee's calculations for the margin to rupture during a LOCA, which accounted for LTA-specific factors that could affect the propensity for rupture, the NRC staff finds that the reinsertion of high burnup LTA U72Y for further irradiation during Cycle 25 does not pose undue risk associated with FFRD. The NRC staff notes that its acceptability determination for the reinsertion of LTA U72Y with respect to FFRD concerns is specifically based on the plant-specific calculations and justifications provided by the licensee and is not applicable beyond the scope of this LAR.

Based on the assessments and analyses the licensee presented in the LAR and the NRC staff's technical evaluations presented above, the NRC staff finds that the licensee will continue to satisfy GDC 35, 10 CFR 50.46, and appendix K to 10 CFR part 50 for acceptable ECCS performance. Further, through the LAR, the licensee has demonstrated that it will continue to meet the radiation dose limits specified in 10 CFR 50.67. Through the evaluations performed in the LAR and the demonstration of applicability of existing methods from the AOR, the NRC staff finds the licensee will continue to meet the 10 CFR 50.46(b)(4) criteria for maintaining a coolable core geometry.

Finally, the staff finds that the proposed changes to TS 2.1.1.3 adequately describe the required reactor core safety limits while U72Y is reinserted in the core for Cycle 25 and, are therefore, sufficient to meet the requirements of 10 CFR 50.36(c)(1). Furthermore, the staff finds that the LTA and the conditions under which the LTA may be placed in the core for Cycle 25 are adequately described in the proposed revision to TS 4.2.1 in that they describe how the fuel may be loaded in the core and the applicable fuel limits. Therefore, the staff finds that the proposed changes are sufficient to meet the requirements of 10 CFR 50.36(c)(4). Based on the above, the NRC staff concludes that TS 2.1.1.3 and TS 4.2.1, as amended by the proposed change, will continue to meet the requirements of 10 CFR 50.36, and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments on April 28, 2023. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (87 FR 74668). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Constellation Energy Generation, LLC letter to NRC dated August 31, 2022, "Byron Station Unit 2 License Amendment Request to Reinsert an Accident Tolerant Fuel Lead Test Assembly," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22243A094)).

2. Constellation Energy Generation, LLC letter to NRC dated August February 27, 2022, "Response to Request for Additional Information for the Byron Proposal to Reinsert an Accident Tolerant Fuel Lead Test Assembly," (ML23058A147).

3. Byron/Braidwood Stations, Units 1 and 2, Updated Final Safety Analysis Report, Rev. 18, December 2020 (ML21008A380).

4. NRC Letter to Nuclear Energy Institute dated June 2019, "Final LTA Guidance Letter. Clarification of Regulatory Path for Lead Test Assemblies," (ML18323A169).

5. NRC Letter to Exelon Generation Company, LLC dated June 2016, "Byron Station Unit Nos. 1 and 2, and Braidwood Station, Units 1 and 2 – Exemption from the Requirements of 10 CFR Part 50, Section 50.46 and Appendix K to allow the Use of Optimized Zirlo Clad Fuel Rods (CAC Nos. MF7399, MF7400, MF7401, MF7402. June 2016)," (ML16125A538).

6. NRC Letter to Constellation Energy Generation, LLC dated April 2019, "Byron Station – Units 2 Issuance of Amendment No. 207 Regarding Use of Accident Tolerant Fuel Lead Test Assemblies," (ML19038A017).

7. WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstruction Evaluation Methodology," July 2093 (ML093630008, Proprietary).

8. WCAP-12610-P-A, Part 1 of 2, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (ML020430250, Proprietary).
9. RIL 2021-13, "Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup," (ML21313A145).

10. WCAP-18482-P-0, Revision 0, "Westinghouse Advanced Doped Pellet Technology (ADOPT™) Fuel," May 2020 (ML20132A014).

11. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ML003716792).

12. NRC Letter to Exelon Generation Company, LLC dated September 8, 2006, "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Alternative Source Term," (ML062340420).

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Date of Issuance: July 20, 2023

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS NOS. 233 AND 233 RE: TECHNICAL SPECIFICATIONS 2.1.1 AND 4.2.1 TO ALLOW A PREVIOUSLY IRRADIATED ACCIDENT TOLERANT FUEL LEAD TEST ASSEMBLY TO BE FURTHER IRRADIATED IN UNIT NO. 2 (EPID L-2022-LLA-0131) DATED JULY 20, 2023

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