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

April 3, 2019

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING USE OF ACCIDENT TOLERANT FUEL LEAD TEST ASSEMBLIES (EPID L-2018-LLA-0064)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 207 to Renewed Facility Operating License No. NPF-66 for the Byron Station, Unit No. 2. The amendment is in response to your application dated March 8, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18067B431), as supplemented by letters dated July 2, 2018, December 18, 2018, and January 16, 2019 (ADAMS Accession Nos. ML18184A270, ML18352B117, and ML19016A491, respectively).

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

Sincerely,

Joel S. Wiebe

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

Docket No. STN 50-455

Enclosures:
1. Amendment No. 207 to NPF-66
2. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 207
Renewed License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated March 8, 2018, as supplemented by letters dated July 2, 2018, December 18, 2018, and January 16, 2019, 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:

   (1) Technical Specifications

   The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 207, 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 prior to startup with accident tolerant fuel lead test assemblies in the reactor.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: April 3, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 207

RENEWED FACILITY OPERATING LICENSE NO. NPF-66

BYRON STATION, UNIT NO. 2

DOCKET NO. STN 50-455

Replace the following pages of the Renewed Facility Operating License 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.

Remove	Insert
License NPF-66	License NPF-66
Page 3	Page 3
TSs	TSs
4.0 – 1	4.0 – 1
4.0 – 2	4.0 – 2

(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. 207, 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.

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 LTAs containing up to twenty total lead test rods may be placed in the reactor for evaluation. 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.

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium, hafnium, or a mixture of both types.

4.3 Fuel Storage

4.3.1 Criticality

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

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

b. A $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Holtec International Report HI-982094, "Criticality Analysis for Byron/Braidwood Rack Installation Project," Project No. 80944, 1998;

c. A nominal 10.888 inch north-south and 10.574 inch east-west center to center distance between fuel assemblies placed in Region 1 racks; and

d. A nominal 8.97 inch center to center distance between fuel assemblies placed in Region 2 racks.

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 410 ft, 0 inches.

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2984 fuel assemblies.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 207 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NO. 2

DOCKET NO. STN 50-455

1.0 INTRODUCTION

By letter dated March 8, 2018, as supplemented by letters dated July 2, 2018, December 18, 2018, and January 16, 2019, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18067A431, ML18184A270, ML18352B117, and ML19016A491, respectively) Exelon Generation Company, LLC (Exelon, the licensee), requested an amendment to the Byron Station, Unit No. 2, operating license that would allow accident tolerant fuel (ATF) lead test assemblies (LTAs) to be utilized. The proposed amendment would amend Byron, Unit 2, technical specifications (TSs), permitting the use of two such LTAs during refueling cycles 22, 23, and 24.

Exelon is requesting to insert two LTAs developed by Westinghouse. These LTAs will be based on the Westinghouse VANTAGE+ Optimized Fuel Assembly (OFA) design which makes up the remainder of the core. The licensee proposed to insert a total of up to 20 lead test rods (LTRs) of three different types between the two LTAs. These rods contain a mixture of three new materials: uranium silicide (U_3Si_2) fuel pellets, ADOPT™ doped uranium dioxide (UO_2) fuel pellets, and coated Optimized ZIRLO™ cladding. The combinations of these technologies proposed for insertion into the reactor are discussed in greater detail in the Technical Evaluation section of this safety evaluation (SE).

Following the second supplement to the license amendment request (LAR), dated December 18, 2018, the U.S. Nuclear Regulatory Commission (NRC or Commission) staff conducted a regulatory audit[1] in order to better understand the issues that prompted the second supplement and to evaluate what additional information, if any, needed to be placed on the docket. The audit is documented in the audit plan, dated December 31, 2018, and the audit report, dated January 15, 2019 (ADAMS Accession Nos. ML19016A235 and ML19017A064, respectively). The licensee addressed open items resulting from the audit and a resulting request for additional information, dated January 11, 2019 (ADAMS Accession No. ML19011A345) in the third supplement to the LAR, dated January 16, 2019.

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[1] A regulatory audit is a planned, license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. A regulatory audit is conducted with the intent to gain understanding, to verify information, and/or to identify information that will require docketing to support the basis of the licensing or regulatory decision.

The supplements dated July 2, 2018, and December 18, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 6, 2018 (83 FR 55573). The supplement dated January 16, 2019, changed the scope of the application as originally noticed by eliminating the license condition and requesting a change to TS 4.2.1 to describe the LTAs and to indicate that the LTAs will be nonlimiting under steady state conditions and will comply with the fuel limits in the core operating limits report (COLR) and the TSs under all conditions. The change in scope and the updated proposed significant hazards consideration was published in the *Federal Register* on February 1, 2019 (84 FR 1240).

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulatory requirements and guidance in its review of the licensee's application.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67, *Accident source term* [AST], states 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, and (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.

10 CFR Part 50, Appendix A, *General Design Criteria for Nuclear Power Plants [GDC], 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.

Regulatory Guide (RG) 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Revision 0, July 2000 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 Station, Unit Nos. 1 and 2, respectively, (see ADAMS Accession No. ML062340420). The amendments fully implemented an AST, pursuant to 10 CFR 50.67 "Accident source term."

Section 50.36 of 10 CFR, "Technical specifications," specifically 10 CFR 50.36(c)(4), "*Design Features*," which requires licensees to include TSs for certain design features which, if altered or modified, would have a significant effect on safety and are not covered in other categories described in 10 CFR 50.36. The Byron, Unit 2, TS have an existing provision in TS 4.2.1 describing the fuel assemblies that may be loaded into the reactor core, including LTAs.

10 CFR 50, Appendix A, *GDC, Criterion 10 – Reactor Design*, states that:

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

10 CFR 50.46(a), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," states in part that:

> Each boiling or pressurized light-water reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section.

By NRC letter dated June 20, 2016 (ADAMS Accession No. ML16126A032), the licensee was granted an exemption to 10 CFR 50.46, which allowed the acceptance criteria of 10 CFR 50.46 to be applied to fuel assembly designs using the Optimized ZIRLO™ fuel rod cladding material.

The NRC staff also considered the guidance in NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition," Section 4.2, Revision 2, "Fuel System Design" (ADAMS Accession No. ML052340660).

3.0 TECHNICAL EVALUATION

3.1 Current Licensing Basis

The licensee's licensing basis includes 193 fuel assemblies. These assemblies have all been of the VANTAGE+ OFA design, with a mixture of ZIRLO™ and Optimized ZIRLO™ cladding. Each fuel assembly consists of 264 fuel rods in a 17x17 array; these assemblies are sometimes referred to as "17 OFA" fuel.

The VANTAGE+ fuel rods have UO_2 pellets contained within the cladding, which is sealed and welded at the ends. Some rods have a zirconium diboride integral fuel burnable absorber (IFBA) added to the fuel pellet.

3.2 Description of LTAs

By letter dated March 8, 2018, the licensee requests approval of the use of the following two LTAs containing a total of up to 20 LTRs. By letter dated July 2, 2018, the licensee described these LTAs as follows:

> LTA #1 is a Westinghouse VANTAGE+ Optimized Fuel Assembly design containing:
>
> - Up to four rods with U_3Si_2 pellets and Optimized ZIRLO™ cladding. The U_3Si_2 pellets will be enclosed in a sealed clad segment with the segment positioned between two solid Zircaloy bars.
> - Up to four rods with standard UO_2 pellets and coated Optimized ZIRLO™ cladding
> - All other rods in LTA #1 will have standard UO_2 pellets and standard Optimized ZIRLO™ cladding
>
> LTA #2 is a Westinghouse VANTAGE+ Optimized Fuel Assembly design and contains:
>
> - Up to eight rods with standard UO_2 pellets and coated Optimized ZIRLO™ cladding
> - Up to four rods with Westinghouse ADOPT™ UO_2 pellets and coated Optimized ZIRLO™ cladding
> - All other rods in LTA #2 will have standard UO_2 pellets and standard Optimized ZIRLO™ cladding
>
> There are no other differences compared to the existing resident VANTAGE+ fuel assembly design.

The licensee further states that the LTRs will not contain any IFBA.

ADOPT™ fuel pellets have small quantities of dopants added during sintering to increase the grain size within the ceramic fuel pellet. This is done to increase fuel density and may have other benefits.

Coated Optimized ZIRLO™ cladding is the same as is in the non-LTA cladding loaded into Byron, Unit 2, but has a thin coating (~25 micron) of chrome applied to the exterior surface of the cladding. This is to reduce corrosion of the Optimized ZIRLO™.

The U_3Si_2 pellets are being explored as a potential replacement for UO_2 pellets. The unirradiated properties of the fuel show increased uranium density and thermal conductivity, and early irradiation testing indicates that fission gas release may also be improved. The pellets for these LTRs are being produced by the U.S. Department of Energy's Idaho National Laboratory (INL).

3.3 <u>Mechanical Design Methodology</u>

In its letter dated March 8, 2018, the licensee states that Westinghouse will evaluate the mechanical design impacts of the new materials on the LTA fuel assemblies. No component changes or changes to the fuel assembly design are expected. The licensee also states that the most significant impacts from the coated cladding are expected to be minor changes in cladding outer diameter, diameter growth and variation, and surface roughness.

The U_3Si_2 pellets and rod segments will have an insignificant impact on rod and assembly weight. In its letter dated March 8, 2018, the licensee states that these pellets will undergo significant out-of-pile (not in a reactor) testing prior to insertion of the LTAs. The pellets are also being irradiated at the INL Advanced Test Reactor. Because these in-pile (in a reactor) programs are limited and have not yet achieved full burnup, the following steps have been taken to reduce the risk associated with U_3Si_2 fuel insertion in the LTAs:

1) There will be a very limited quantity of U_3Si_2 in the LTAs. A total of up to four rods will each contain a section of U_3Si_2 fuel pellets that is approximately 1 foot long, with the remainder of the rod consisting of a solid zircaloy bar.
2) The section of fuel containing U_3Si_2 will be positioned between spacer grids. This should eliminate fuel failure due to grid-to-rod fretting and greatly reduce the chance of failure due to debris (which often occurs after debris is caught in a spacer grid and vibrates adjacent to the fuel rod).
3) The segment containing the U_3Si_2 fuel pellets will have a thicker clad than the other rods in the core.

The NRC staff finds that the very small quantity of U_3Si_2, coupled with the increased clad thickness and segmented rod location will reduce the risk and potential consequences of fuel clad failure. It should also be noted that the licensee states that the existing primary coolant activity monitoring procedures will be able to detect a leak in the U_3Si_2-fueled lead test rods as it does for UO_2 fuel. Based on the above, the NRC staff concludes that there is reasonable assurance that the U_3Si_2 fuel insertion in the LTA will not adversely affect safety.

The licensee states that coated Optimized ZIRLO™ cladding is undergoing characterization out of pile and thus far compares favorably with Optimized ZIRLO™. In-pile testing in test reactors of coated Optimized ZIRLO™ is ongoing and will be performed in parallel with the LTA irradiation.

Westinghouse has operating experience using ADOPT™ fuel pellets outside of the United States, with irradiation of over 1800 fuel assemblies inserted into boiling-water reactors. Westinghouse also has experience with ADOPT™ in European pressurized-water reactors. The licensee states that fuel performance for these pellets is well understood and the pellets will have negligible impact on the mechanical design of the assembly.

The assembly and rod mechanical design will be based on a previously approved Westinghouse assembly design (the 17 OFA assembly). All fuel rod design criteria are to be evaluated to confirm sufficient margins exist. By letter dated December 18, 2018, the licensee states that both the ADOPT™ fuel pellets and the coated cladding will be evaluated using the licensed PAD fuel performance code, but with input parameters changed to reflect the best state of knowledge for the new technologies. Due to the limited number of lead test rods, the NRC staff finds it acceptable to extend the use of the approved licensing code by modifying the inputs to model these LTRs within the limited scope of this LTA program.

Analysis of the U_3Si_2 fuel will be performed using a version of PAD (performance code) named PAD-ATF, which has been modified to include the properties of this new fuel pellet. LTA irradiation programs and subsequent post-irradiation examinations are needed to collect the data that is included in topical reports and used to evaluate the acceptability of new codes and methods. Thus, for the purposes of the irradiation of the U_3Si_2 LTRs, the NRC staff finds the use of an approved code with modifications to add new material properties (i.e., PAD-ATF) acceptable, as fully approved codes are unavailable and the quantity and placement of the LTAs provide reasonable assurance that that there is not a significant risk to the public. This is not to be considered approval of PAD-ATF for anything beyond this specific LTA program.

3.4 Seismic

In its letter dated March 8, 2018, the licensee states that the impact of the LTAs on the seismic evaluation will be negligible due to the small change in weight of the fuel rods. Westinghouse will evaluate grid deformation to ensure margins are maintained. The NRC staff finds this acceptable based on the LTA similarity to the existing fuel assemblies.

3.5 Core Physics

In its letter dated July 2, 2018, the licensee states that the primary design criterion for the U_3Si_2 LTRs is to maintain the U_3Si_2 segments' power peaking at least 5 percent below the cycle's core power peaking during steady-state conditions. The design criteria for the ADOPT™ and coated clad LTRs will be to maintain power peaking at least 2 percent below the cycle's core maximum power peaking during steady-state conditions. The LTA will also be below the average enrichment for the reload and will not be designed to lead the core in power peaking at any time during normal operation. As discussed in Section 3.7 of this safety evaluation, ADOPT™ pellets may lead the core in certain transients though they remain below the acceptable limits for the co-resident UO_2 pellets. Based on the above and that the pellets will remain within the technical specifications and COLR fuel limits, the NRC staff finds that 10 CFR 50, Appendix A, GDC 10, is met.

3.6 Loss-of-Coolant Accidents (LOCA)

In its March 8, 2018, letter the licensee states that the LTAs will have an insignificant impact on the consequences of a postulated LOCA due to the small number of LTRs. The licensee further states that the U_3Si_2 fuel stored energy should be lower than UO_2 due to the enhanced thermal conductivity, and the pre-transient corrosion and LOCA oxidation performance of the coated cladding is expected to be improved. Because the behavior of the LTRs is not expected to vary significantly from that of the co-resident fuel, and the limited number of LTRs in the core limits the risk from unexpected behavior, the NRC staff finds this acceptable.

The proposed revision to TS 4.2.1 requires the LTAs to be nonlimiting under steady-state conditions. As discussed in the Byron Updated Final Safety Analysis Report (UFSAR), Section 15.6.5 (ADAMS Accession No. ML17086A599), the LOCA analysis is initiated from steady-state conditions. Based on these facts, the limiting fuel assemblies in the reactor core will continue to be those with UO_2 fuel in zircaloy, ZIRLO®, or Optimized ZIRLO™ cladding for the purposes of the ECCS performance evaluation. In other words, the LTAs will not be limiting for the purposes of the ECCS performance evaluation. The requirements of 10 CFR 50.46, therefore, continue to apply to the Byron Station, Unit 2, reactor core containing a limited quantity of nonlimiting LTAs, without the need for an exemption. The requirement to locate the LTAs in core regions which are nonlimiting under steady state conditions, the primary design criteria discussed in Section 3.5 of this SE, and expected performance of the LTAs, discussed elsewhere in the SE (including sections 3.7, 3.8, and 3.9) provide reasonable assurance that the LTAs remain nonlimiting under steady-state conditions. The licensee must therefore continue to comply with the 10 CFR 50.46 ECCS acceptance criteria. The ECCS performance evaluation criteria apply to the limiting or worst case rod, fuel assembly, or core area, as appropriate for the analysis being evaluated. Based on the above, the NRC staff finds that 10 CFR 50.46 is met.

3.7 Non-LOCA Events

Events dependent on core-average effects are not expected to experience any significant impact due to the small number of LTRs in the core. In its letter dated March 8, 2018, the licensee states that it will evaluate non-LOCA events whose behavior is dominated by local effects, such as the hot rod, hot channel, or hot spot, to ensure that the applicable Updated Final Safety Analysis Report (UFSAR) sections will remain valid. The NRC staff finds this acceptable, because the limited number of LTAs and steady-state nonlimiting locations limit any possible negative effect on these events from unexpected behavior of the LTRs.

In its letter dated December 18, 2018, the licensee notified the NRC that during the evaluation of the boron dilution transient one case resulted in a higher nodal linear heat generation rate (LHGR) in an ADOPT™ LTR than the co-resident fuel. The NRC staff performed a regulatory audit to better understand the context of this result and how it was derived. In its letter dated December 18, 2018, the licensee states that the co-resident fuel utilizes lower-enriched axial blanket regions at the very top and bottom of each fuel rod, while the ADOPT™ fuel rods do not. This resulted in the highest nodal LHGR being identified in the lower part of the ADOPT™ fuel rod during one of many cases for the boron dilution transient with rods in manual control. This transient is included in the Byron, Unit 2, UFSAR, Chapter 15, analyses and is evaluated as part of the reload safety analysis checklist.

As discussed in Section 3.2 of this SE, Westinghouse has prior experience with ADOPT™ fuel pellets. Westinghouse stated that the ADOPT™ pellets that lead the core during this transient are still well below the fuel centerline melt temperature limit in TS Safety Limit 2.1.1.3. In its letter dated January 16, 2019, the licensee states that the ADOPT™ pellets have essentially the same heat capacity, thermal diffusivity, thermal expansion coefficient, and melting temperature as standard UO_2 pellets. This indicates that it is appropriate to apply the TS safety limit for fuel melt to ADOPT™ pellets.

Because of the very limited quantity of ADOPT™ LTRs, the ability of the licensee to detect any fuel failures, and the relative confidence gained through operating experience in Europe, that the performance of the pellets is not degraded compared to the standard UO_2 product, the NRC staff finds it acceptable to insert ADOPT™ LTRs in positions where they may lead the core during analyzed transients. These rods shall be analyzed using the current state of knowledge to inform input parameters to the licensed codes and methods, and may not lead the core during normal operation or exceed any safety limits that apply to the co-resident UO_2 fuel pellets.

3.8 Thermal-Hydraulic

In its letter dated March 8, 2018, the licensee states that Westinghouse will perform the thermal-hydraulic design evaluations using existing methods. These evaluations will ensure margin to departure from nucleate boiling. Critical heat flux will be tested experimentally to ensure these methods remain applicable to the LTRs, and that no margin is lost due to the coating. Surface roughness of the coated cladding will also be evaluated to confirm that it is similar to uncoated cladding and thus ensure hydraulic compatibility. Similarly, assessments will be performed to validate that the LTR thermal-hydraulic reload design evaluations remain bounded by current analyses, and that there are no adverse effects on the thermal-hydraulic design of the remainder of the core due to the LTAs.

There is little change with respect to thermal hydraulics between the LTAs and standard fuel assemblies, with the exception of the coated rods, as standard structural components will be utilized and the assembly geometry will be unaltered. The expected effect on the thermal-hydraulic characteristics of the LTAs from the thin coating of chrome is expected to be negligible. Based on the facts that the thermal hydraulic design of the LTAs is essentially the same as the approved co-resident fuel, that the assemblies will continue to be analyzed using approved codes and methods, and that the limited number of rods in the core reduces their impact, the NRC staff finds this to be acceptable.

3.9 Fuel Rod Design

While fuel performance is expected to improve for each of the material changes in the LTAs, there remains uncertainty. To mitigate this, Westinghouse will analyze the LTAs using the previously approved PAD code, as well as PAD-ATF, a developmental version of the PAD code that includes (not yet approved) fuel performance models developed from data collected from various sources. Given the very limited number of LTAs and the conservatism from both the additional measures against fuel failure discussed in Section 3.2 and the steady state nonlimiting locations within the core, the NRC staff finds the modifications to previously approved methods acceptable.

Based on the evaluation in sections 3.2 through 3.9, above, the NRC staff finds that the requirements of 10 CFR 50, Appendix A, GDC 10 is met and that the guidance in NUREG-0800, Section 4.2, is met.

3.10 Fuel Handling, Storage, and Shipping

The licensee states that the LTAs are not expected to impact any criticality analyses for the spent fuel pool, new fuel vault, Traveler™ shipping package, or fuel handling equipment. It is further stated that evaluations of the LTAs with U_3Si_2 rods have been performed for the spent fuel pool and new fuel vault and the resulting impact to reactivity was found to be negligible.

Additional analyses were performed on the reactivity trajectory that found little impact for spent fuel pool racks or dry storage. Based on the little impact expected from the limited number of LTRs and that the licensee has explicitly accounted for the changes in the pertinent analyses, the NRC staff finds this to be acceptable.

The Traveler™ container has a separate license under 10 CFR Part 71, and the appropriateness of using the container to transport the LTAs has not been evaluated in this review.

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

In its letter dated March 8, 2018, the licensee states that the BEACON™ core monitoring system will be unaffected by the LTAs. The LTAs will be placed to have a negligible effect on the measurements of the power distribution monitoring system. Nuclear data libraries for both new fuel pellets will be developed to model these rods explicitly. NRC staff finds this acceptable, based on the limited number of LTRs in the core, coupled with the explicit modeling of new materials, which will ensure that the power distribution in the core can be monitored.

3.12 Source Term

The NRC staff reviewed the fuel system design basis and the assumptions, inputs, and methods used by the licensee to develop accident-specific source terms for applicable DBA radiological dose consequence analyses. The purpose of the review was to determine if the analysis of record remains bounding and if the applicable doses will remain below regulatory limits. The review also considered any corresponding TS that are dependent on the fuel parameters to determine if the TS bound operation of LTAs in nonlimiting core locations.

The current Byron, Unit 2, licensing basis for the radiological consequences is discussed in Section 15, "Accident Analysis," in the UFSAR. The analyses are based upon the AST methodology prescribed in RG 1.183 for the offsite- and onsite doses at the EAB, LPZ, and CR locations. Many of the accidents involve failure or malfunction of systems which could affect the fuel in the reactor vessel. The most severe would involve the release of large quantities of fission products (herein referred to as the source term). The staff reviewed five applicable bounding DBA radiological consequence analyses, focusing on the modeling assumptions and accident-specific source terms for each. These DBAs are not expected to take place, but are postulated because their consequences would include the potential release of significant amounts of radioactive material. Therefore, they represent limiting design cases for which the facility must be designed so that these events would not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of the guideline values of 10 CFR 50.67 for AST based analyses using RG 1.183 methodology. These accidents involve failure or malfunction of the reactor, reactor cooling system, steam system, or turbine generator, and include:

- loss-of-coolant accident (LOCA), UFSAR Section 15.6.5;
- control rod ejection accident (CREA), UFSAR Section 15.4.8.3;
- locked rotor accident (LRA), UFSAR Section 15.3.3;
- main steam line break (MSLB), UFSAR Sections 15.1.5-6; and,
- steam generator tube rupture (SGTR), UFSAR Section 15.6.3.

See Section 3.5, “Core Physics,” and Section 3.13, “Technical Specifications,” in this SE for a description of the expected effects of the LTAs in the core. Consistent with the guidance of RG 1.183, Position 3.1, the licensee used ORIGEN 2.1 methodology to determine the reactor core inventory of fission products based on the fuel system design discussed in the UFSAR, Section 4.2, "Fuel System Design" (ADAMS Accession No. ML16357A520). The core consists of 193 fuel assemblies composed of two regions of Westinghouse VANTAGE+ OFAs, one region with ZIRLO™ cladding and the other with Optimized ZIRLO™ cladding. Each fuel assembly consists of 264 fuel rods arranged in a 17x17 array. There are a total of 50,952 fuel rods in the core.

The reactor core inventory available for release associated with each DBA is computed at a reactor power level of 3,658.3 MW (megawatt), based on the current licensed values for fuel enrichment, burnup, and an additional 2 percent uncertainty factor to account for emergency core cooling system (ECCS) evaluation uncertainty. Consistent with the guidance of RG 1.183, Position 3.2, for the DBA LOCA, all fuel assemblies in the reactor core are assumed to be affected during the gap release and early in-vessel damage phases. For the non-LOCA events, the fractional fission product inventory is assumed as a gap release by each damaged fuel rod in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

The two LTAs will contain up to a combined total of 20 LTRs. The specific composition of the two LTAs are described in Section 3.2, above.

The combined total of 20 LTRs represents 0.039 percent of the total reactor core inventory. Of these 20 LTRs, four rods contain approximately 1 linear foot each of U_3Si_2 pellets; representing 0.008 percent of the reactor core inventory. The 12 fuel rods with standard UO_2 fuel pellets coated with Optimized ZIRLO™ cladding represent 0.024 percent of the reactor core inventory. The four fuel rods containing ADOPT™ fuel pellets represent 0.008 percent of the reactor core inventory.

The LOCA DBA radiological consequence analysis is consistent with the release fractions in RG 1.183, Table 2, and transport fractions per RG 1.183, Appendix A. The analysis assumes the amount of radionuclides that would be released from the reactor core inventory is: (1) proportional to the amount of radionuclides in the reactor core; and, (2) not significantly affected by the gap-release fraction. The gap-release fraction is a small contributor to the amount of radionuclides available for release when the fuel is severely damaged. Any increase in the amount of some longer-lived radionuclides available for release from the LTAs is: (1) expected to be small; and, (2) is not expected to result in a significant increase in the overall core inventory of radionuclides. Therefore, there would be little significant increase in the previously calculated dose from a LOCA and the dose would remain below regulatory limits.

The CREA radiological dose analysis is consistent with the release fractions per RG 1.183, Table 3, and transport fractions per RG 1.183, Appendix H. The accident assumes the sudden rod ejection and localized temperature spike damages 10 percent of the reactor core fuel rods. Only 2.5 percent of the damaged core fuel rods release melted fuel activity. Therefore, the source term available for release is associated with this fraction of melted fuel and the fraction of core activity existing in the gap. The damaged fuel is assumed to have operated at a limiting core location with the maximum core radial peaking factor of 1.7 applied to the calculated fission product inventory. As discussed above, to be consistent with TS 4.2.1, the licensee states that the core design will ensure that the LTRs within the LTAs, and the LTAs in the core, will be located in nonlimiting core locations for steady state operation and in all cases will comply with

TS and the COLR limits. Therefore, there would be an insignificant increase in the previously calculated dose and results would remain below regulatory limits.

The LRA radiological dose analysis is consistent with the release fractions per RG 1.183, Table 3, and transport fractions per RG 1.183, Appendix G. It is assumed that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. It is assumed conservatively that, as a result of the LRA, 2 percent of the fuel rods in the core undergo sufficient clad damage to result in the release of their gap activity. The design basis of this accident assumes that no fuel melt is postulated to occur. Therefore, no melted fuel enters the reactor coolant system (RCS). As such, two bounding cases are considered. The first case assumes the source term available for release is composed of the: (1) associated 2 percent fraction of damaged fuel caused by the event, (2) fraction of core activity existing in the gap, (3) iodine in the RCS due to a design basis pre-accident 60 μCi/gm (microcurie per gram) dose equivalent iodine (DEI) (I-131) spike, and (4) an additional noble gas activity associated with 1 percent fuel defects within the RCS. The second case includes the additional source activity consisting of the 0.1 μCi/gm DEI equilibrium secondary coolant activity concentration limit from the TS. The LTRs represent a very small fraction of the total reactor core inventory, 0.039 percent. Any increase in the amount of some longer-lived radionuclides available for release from the LTAs is: (1) expected to be small and, (2) not expected to result in a significant increase in the overall core inventory of radionuclides. Therefore, there would be an insignificant increase in the previously calculated dose from a LRA event and the results would remain below regulatory limits.

The MSLB and SGTR radiological dose analyses are consistent with the release fractions per RG 1.183, Table 3, and transport fractions per RG 1.183, Appendices E and F, respectively. The design basis assumes no fuel damage for either event. The source terms are defined by the TS activity release rates from a maximum failed fuel fraction assumed during operation, which are characterized by the equilibrium 1.0 μCi/gm DEI activity concentration in the primary RCS. Additionally, no change is being requested by the licensee in the Byron, Unit 2, TS pertaining to the allowed RCS activity concentrations. The maximum RCS activity is regulated through TSs that are dependent on reactor core power, which are also being not being changed. Therefore, the gap-release fraction does not significantly affect the amount of fission products available for release during either event. The noble gas inventory in the RCS is based on operation with a conservative worst-case 1 percent core fuel defects which is derived from the reactor core inventory. Since no fuel damage is assumed for either accident, only the coolant activity with iodine spiking and noble gas isotopes are modeled to contribute to dose. As such, two cases of iodine spiking are analyzed, per regulatory guidance. In the first case, a 60 μCi/gm pre-accident iodine spike is modeled for both the MSLB and SGTR. This 60 μCi/gm spike is consistent with the TS operational RCS activity concentration limit for an assumed spike. In this scenario, it is assumed that all of the spike activity is homogeneously mixed in the primary coolant, prior to accident initiation. The second case involves an accident initiated iodine spike that occurs concurrently with the release of fluid from the primary and secondary coolant systems. For the MSLB and SGTR, regulatory guidance specifies that this spike should result in a release rate from the operating limit defective fuel fraction that is respectively 500 and 335 times the normal rate. Operation of a limited number of LTAs in nonlimiting core locations is not expected to significantly increase the overall RCS inventory of radionuclides. Since no fuel damage is expected to occur as a result of the MSLB and SGTR events, there will be an insignificant increase in the previously calculated dose and the results would remain below regulatory limits.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to develop accident-specific source terms for DBA radiological dose consequence analyses. The NRC staff finds that Byron, Unit 2, will continue to meet the applicable dose limits while operating with the limited number of LTAs in nonlimiting core locations for steady state conditions and complying with all COLR limits during transients. The NRC staff further finds reasonable assurance that Byron, Unit 2, as modified by this amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed utilization of 2 LTAs is acceptable with respect to the assumptions, inputs, and methods used by the licensee to develop accident-specific source terms for DBA radiological dose consequence analyses.

Based on the above, the NRC staff finds 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.

3.13 Technical Specifications

The licensee has proposed the following language be added to the TS following the existing text in section 4.2.1, "Fuel Assemblies":

> During Unit 2 Cycles 22, 23, and 24, two LTAs containing up to twenty total lead test rods may be placed in the reactor for evaluation. 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.

Based on the evaluation in Section 3 of this SE, the NRC staff finds that the utilization of these LTAs is acceptable, and this TS change adequately describes the LTAs and the conditions under which each type of LTA may or may not be limiting. Based on the above, the NRC staff concludes TS 4.2.1, as amended by the proposed change, will continue to meet the requirements of 10 CFR 50.36(c)(4).

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 February 6, 2019. The State official had no comments.

5.0 PUBLIC COMMENT

On November 6, 2018, the NRC staff published in the *Federal Register* a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," associated with the proposed LAR (83 FR 55573). In accordance with the requirements in 10 CFR 50.91, the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. On February 1, 2019, the NRC staff published in the *Federal Register* an updated "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," associated with the proposed amendment request (84 FR 1240) in response to the licensee's letter dated January 16, 2019, which revised the scope of the application. In accordance with the requirements in 10 CFR 50.91, the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination.

Public comments were received regarding the proposed amendment on November 28, 2018 (ADAMS Accession No. ML18333A046). The public comments do not specifically pertain to the proposed NSHC determination. The comments primarily pertain to the NRC staff's current effort to establish a consistent regulatory path for utilization of lead test assemblies. (See *Federal Register* Notice (FRN) dated June 7, 2018 (83 FR 26503), as supplemented by the FRN dated July 2, 2018 (83 FR 30989) in which the NRC solicited public comments on a draft letter to the Nuclear Energy Institute "clarifying the regulatory paths for the use of lead test assemblies (LTAs)" (ADAMS Accession No. ML18100A045)). The guidance provided in the draft letter to the Nuclear Energy Institute is not within the scope of this safety evaluation; however, the NRC staff has addressed the comments within the context of this amendment. A summary of the comments and the NRC staff responses are addressed below.

5.1 Need for an Exemption

Summary of Comment

Byron Station, Unit 2, needs an exemption because not requiring an exemption is a new interpretation of 10 CFR 50.46 that is not supported by the language and history of the rule.

Response to Comment

Based on the discussion in section 3.6 of this SE, the NRC staff finds that 10 CFR 50.46 is met and an exemption is not required.

The NRC is following precedent set in 2017 with Southern Nuclear Operating Company with regards to the Edwin I. Hatch Nuclear Plant loading LTAs with new materials without an exemption; this decision was communicated during a phone call and documented in a publicly available summary of that call (ADAMS Accession No. ML18039A979).

5.2 License Authority

Summary of Comment

The commenter stated that NRC does not have the authority to grant this license amendment without an exemption from 10 CFR 50.46 and described an exemption that was issued in 2018 for Beaver Valley Power Station, Units 1 and 2 (ADAMS Accession No. ML17313A554).

Response to Comment

The NRC staff does not believe an exemption from 10 CFR 50.46 is necessary in this case, for reasons described above. The NRC staff notes that the referenced Beaver Valley exemption was not related to an LTA campaign, but instead supported a license amendment seeking the use of Optimized ZIRLO™ in reload quantities at the plant and is therefore not relevant to this amendment. In addition, the NRC staff's decision to issue exemptions that licensees have requested in past circumstances does not alter the NRC's authority to issue the license amendment requested by the licensee in this case.

5.3 Proposed Amendment Needs to Revise TS 4.2.1

Summary of Comment

The commenter noted that the existing TS for Byron, Unit 2, state at TS 4.2.1 that "[e]ach 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 (UO2) as fuel material." The commenter stated that this license amendment needs to change TS 4.2.1 to allow the use of alternative cladding and fuel materials.

Response to Comment

On January 16, 2019, after this comment was submitted, the licensee submitted a supplement to its license amendment application. This supplement requests that TS 4.2.1 be revised as described in Section 3.13 above. The NRC staff believes that additional TS changes are not necessary for this amendment because the revised TS 4.2.1 clearly specifies the type of cladding and fuel pellets allowed to be contained in the LTAs.

5.4 Congressional Review Act (CRA) Considerations

Summary of Comment

The commenter stated that the draft guidance letter on LTAs constitutes a "rule" under the Congressional Review Act (5 U.S.C. § 801) and should be processed under the NRC's procedures related to that Act. Therefore, the commenter argued that the NRC should not implement the guidance letter before it is finalized.

Response to Comment

While the draft guidance letter on LTAs is outside the scope of this license amendment review, the NRC agrees that that document has not been finalized. The NRC staff's evaluation of this license amendment request did not rely on the draft guidance letter.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 previously issued a proposed finding on June 5, 2018 (83 FR 26104), and an updated proposed finding on February 1, 2019 (84 FR 1240) that the amendments involve no significant hazards consideration. The public comments received on the proposed finding, which are discussed in Section 5.0 above, did not specifically pertain to the proposed no significant hazards consideration determination. 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.

7.0 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.

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Date of issuance: April 3, 2019

SUBJECT: 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

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