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

July 17, 2019

Mr. J. Ed Burchfield, Jr.
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 413, 415, AND 414 REGARDING THE UPDATED FINAL SAFETY ANALYSIS REPORT SECTION FOR FISSION GAS GAP RELEASE RATES (EPID NO. L-2018-LLA-0300)

Dear Mr. Burchfield:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment Nos. 413, 415, and 414 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments revise the Updated Final Safety Analysis Report (UFSAR) in response to the application from Duke Energy Carolinas, LLC via letter RA-18-0136 dated November 1, 2018, as supplemented by letter RA-19-0134 dated March 7, 2019.

The amendments revise the dose consequences for the facility as described in the UFSAR to provide fission gas gap release fractions for high-burnup fuel rods that exceed the linear heat generation rate limit detailed in Regulatory Guide (RG) 1.183, Table 3, Footnote 11. The amendments allow a higher bounding rod power history and the removal of a restriction on the number of rods per assembly that can exceed the rod power burnup criteria of Footnote 11 in RG 1.183.

The staff's safety evaluation of the amendments is enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Michael Mahoney, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 413 to DPR-38
2. Amendment No. 415 to DPR-47
3. Amendment No. 414 to DPR-55
4. Safety Evaluation

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SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 413, 415, AND 414 REGARDING THE UPDATED FINAL SAFETY ANALYSIS REPORT SECTION FOR FISSION GAS GAP RELEASE RATES (EPID NO. L-2018-LLA-0300) DATED JULY 17, 2019

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ADAMS Accession No.: ML19183A317 ***By E-mail +By Memo**

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 413
Renewed License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. DPR-38, filed by Duke Energy Carolinas, LLC (the licensee), dated November 1, 2018, as supplemented by letter dated March 7, 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 as 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 set forth in 10 CFR Chapter I;

   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, by Amendment No. 413, Renewed Facility Operating License No. DPR-38 is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated November 1, 2018, as supplemented by letter dated March 7, 2019. The licensee shall update the UFSAR to incorporate the changes as described in the licensee's application dated November 1, 2018, as supplemented by letter dated March 7, 2019, and the NRC staff's safety evaluation enclosed with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance. The UFSAR changes shall be submitted in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e) following the implementation period.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: July 17, 2019

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 415
Renewed License No. DPR-47

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

    A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. DPR-47, filed by Duke Energy Carolinas, LLC (the licensee), dated November 1, 2018, as supplemented by letter dated March 7, 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 as 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 set forth in 10 CFR Chapter I;

    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, by Amendment No. 415, Renewed Facility Operating License No. DPR-47 is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated November 1, 2018, as supplemented by letter dated March 7, 2019. The licensee shall update the UFSAR to incorporate the changes as described in the licensee's application November 1, 2018, as supplemented by letter dated March 7, 2019, and the NRC staff's safety evaluation enclosed with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance. The UFSAR changes shall be submitted in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e) following the implementation period.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: July 17, 2019

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 414
Renewed License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility), Renewed Facility Operating License No. DPR-55, filed by Duke Energy Carolinas, LLC (the licensee), dated November 1, 2018, as supplemented by letter dated March 7, 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 as 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 set forth in 10 CFR Chapter I;

   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, by Amendment No. 414, Renewed Facility Operating License No. DPR-55 is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated November 1, 2018, as supplemented by letter dated March 7, 2019. The licensee shall update the UFSAR to incorporate the changes as described in the licensee's application dated November 1, 2018, as supplemented by letter dated March 7, 2019, and the NRC staff's safety evaluation enclosed with this amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance. The UFSAR changes shall be submitted in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e) following the implementation period.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: July 17, 2019

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR

AMENDMENT NO. 413 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 415 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 414 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter RA-18-0136 dated November 1, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18318A320), as supplemented by letter RA-19-0134 dated March 7, 2019 (ADAMS Accession No. ML19066A316), Duke Energy Carolinas, LLC (the licensee) applied for license amendments to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), respectively. The licensee requested to revise the dose consequences for the facility as described in the Updated Final Safety Analysis Report (UFSAR) to: provide fission gas gap release fractions for high-burnup fuel rods that exceed the linear heat generation rate (LHGR) limit detailed in Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792), Table 3, Footnote 11; allow a higher bounding rod power history; and remove a restriction on the number of rods per assembly that can exceed the rod power burnup criteria of Footnote 11 in RG 1.183.

By electronic mail (e-mail) dated February 19, 2019 (ADAMS Accession No. ML19053A562), the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff sent the licensee requests for additional information (RAIs). By letter dated March 7, 2019 (ADAMS Accession No. ML19066A316), the licensee responded to the NRC staff's requests. The licensee's supplement 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* (FR) on January 31, 2019 (84 FR 811).

2.0 REGULATORY EVALUATION

2.1 System Descriptions, Requirements, and Design Bases

An accident source term is the type and amount of radioactive or hazardous material released to the environment (e.g., reactor coolant system or spent fuel pool) following an accident. An accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA) or with accident sequences of lesser consequence but higher probability of occurrence. An alternative source term (AST) is an accident source term that is different from that used in the original design and licensing of the facility and that has been approved for use under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.67, "Accident source term."

Release fraction is the fraction of a particular radionuclide released from the fuel pellet into the void volume of the fuel rod. The void volume is the combined volume of the rod plenum, minus the spring volume, and any gaps between the pellet and the cladding inner diameter. Upon cladding failure, the release (or gap) fraction is able to be released from the fuel rod into the surrounding environment.

RG 1.183 provides guidance on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This RG establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. RG 1.183, Section 3.2, "Release Fractions," Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," lists the fractions of the core inventory assumed to be in the gap for the various radionuclides for non-LOCA events. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. The one footnote for Table 3 (i.e., Footnote 11) states:

> The release fractions listed here have been determined to be acceptable for use with currently approved LWR [light water reactor] fuel with a peak burnup up to 62,000 MWD/MTU [megawatt days per metric ton of Uranium] provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU [gigawatt days per metric ton of Uranium]. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR [boiling water reactor] rod drop accident and the PWR [pressurized water reactor] rod ejection accident, the gap fractions are assumed to be 10% [percent] for iodines and noble gases.

On July 19, 2016 (ADAMS Accession No. ML16159A336), the NRC issued Amendment Nos. 401, 403, and 402 for Oconee 1, 2, and 3, respectively, that revised the UFSAR and approved the use of a new set of fission gas gap release fractions for high burnup fuel rods that exceed the LHGR limit in RG 1.183, Table 3, Footnote 11.

Chapter 15, "Accident Analyses" of the UFSAR details the expected response of the plant to the spectrum of transients and accidents which constitute the design basis events. Section 15.1.10,

"Environmental Consequences Calculation Methodology," describes fission product inventories in the reactor core, fuel pellet clad gap, reactor coolant, steam generators, and secondary side systems. It also discussed the calculation of accident doses. Section 15.11, "Fuel Handling Accidents" describes fuel handling accidents in the spent fuel pool and inside containment, and shipping cask drop accidents, among others. Table 15-1, "Reg. Guide 1.183 Fuel Handling Accident Source Term," describes the fuel assembly gap inventory for fuel handing accidents. Table 15-16, "Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores with Replacement Steam Generators," provides a summary of the offsite dose consequences for transients and accidents.

2.2 Licensee's Proposed Changes

The licensee requested to revise the dose consequences for the facility as described in Chapter 15 of the UFSAR to: provide fission gas gap release fractions for high-burnup fuel rods (i.e., greater than 54 gigawatt days per metric ton of Uranium (GWD/MTU)) that exceed the 6.3-kW/ft LHGR limit detailed in RG 1.183, Table 3, Footnote 11; allow a higher bounding rod power history; and remove a restriction on the number of rods per assembly that can exceed the rod power burnup criteria of Footnote 11 in RG 1.183. In its application, the licensee stated that this request was an update to Amendments 401, 403, and 402 in that the new request applies the American National Standards Institute (ANSI) / American Nuclear Society (ANS) ANSI/ANS-5.4-2011 method exclusively and would allow a higher bounding rod power history and the removal of the restriction in Amendments 401, 403, and 402 on the number of rods (25) per assembly that can exceed the rod power/burnup criteria in Footnote 11.

In Section 15.1.10 of the UFSAR, the licensee proposed the following change (deletions shown as stricken text, additions shown as underlined text):

> *Inventory in the Fuel Pellet Clad Gap:* The fuel pin gap activities were determined using Regulatory Guide 1.183 (Section 15.1, Ref. 35). ~~For non-DNB fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (Reference 46 and 47). A maximum of 25 fuel rods, per fuel assembly, shall be allowed to exceed the rod power/burnup criteria for Footnote 11 in RG 1.183 in accordance with the license amendment request submitted by letter dated July 15, 2015 (Reference 46).~~ For non-DNB fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values (References 46 and 47). The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident. The fuel cycle design also ensures that no fuel rod predicted to experience DNB in any other non-LOCA accidents (e.g. locked rotor accident or rod ejection accident) will have operated beyond the power/burnup criteria of Footnote 11 in Regulatory Guide 1.183 and that the gap fractions used in these non-LOCA accident analyses remain those stated in Table 3 of RG 1.183. ….

In the list of references shown on the last page of Section 15.1 of the UFSAR, the licensee proposed the following changes:

46. ~~Repko, Regis T (Duke Energy) to USNRC, *License Amendment Request Proposing a New Set of Fission Gas Gap Release Fractions for High Burnup Fuel Rods that Exceed the Linear Heat Generation Rate Limit Detailed in Regulatory Guide 1.183, Table 3, Footnote 11*, July 15, 2015.~~ Burchfield, J. E., Jr. (Duke Energy) to USNRC, *License Amendment Request Proposing a Revised Set of Fission Gas Gap Release Fractions for High Burnup Fuel Rods that Exceed the Linear Heat Generation Rate Limit Detailed in Regulatory Guide 1.183, Table 3, Footnote 11; License Amendment Request No. 2018-05*, dated November 1, 2018, as supplemented by letter from Burchfield, J. E., Jr. (Duke Energy) to USNRC, *Duke Energy Response to NRC Request for Additional Information (RAI) Related to Oconee License Amendment Request 2018-05*, dated March 7, 2019.

47. ~~Hall, James R (USNRC) to Repko, Regis T (Duke Energy), *Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3 - Issuance of Amendments Regarding Request to Use an Alternate Fission Gas Gap Release Fraction (CAC NOS. MF6480, MF6481, MF6482, MF6483, MF6484, MF6485, and MF6486)*, July 19, 2016.~~ Mahoney, Michael (USNRC) to Burchfield, J. E., Jr. (Duke Energy), *Oconee Nuclear Station, Units 1, 2, and 3 – Issuance of Amendment Nos. 413, 415, and 414 Regarding the Updated Final Safety Analysis Report Section for Fission Gas Gap Release Rates (EPID NO. L-2018-LLA-0300),* [Date of issuance of amendments]

In Section 15.11.2.1, “Base Case Fuel Handling Accident in Spent Fuel Pool,” of the UFSAR, the licensee proposed the following change:

> The fuel assembly gap inventory is assumed to contain a fission product inventory from a maximum burned fuel assembly at a radial peaking factor of 1.65. The gap fractions used are from Reg. Guide 1.183 and the reactor has been shutdown for 72 hours, which is the minimum time for RCS cooldown, reactor closure head removal, and removal of the first fuel assembly. ~~For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.~~ For fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values. ….

In Section 15.11.2.2, “Base Case Fuel Handling Accident inside Containment,” of the UFSAR, the licensee proposed the following change:

> Using the fuel assembly gap inventory in Table 15-1, and assuming all 208 fuel pins are damaged, the calculated doses are appropriately within the guidelines given in Regulatory Guide 1.183. ~~For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG~~

~~1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.~~ For fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values. ….

In Section 15.11.2.4, "Shipping Cask Drop Accidents," of the UFSAR, the licensee proposed the following change:

10. The fractions of noble gases and iodine in the gaps are shown below. ~~For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals~~ For fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values (Reference 1).

In Section 15.11.3, "References," of the UFSAR, the licensee proposed the following change:

1. DPC Engineering Calculation OSC7738, "Fuel Handling Accidents (FHA) Dose Analysis," dated ~~January 28, 2010~~ September 10, 2018.

In Table 15-1, "Reg. Guide 1.183 Fuel Handling Accident Source Term," of the UFSAR, the licensee proposed the following change to the Note (deletions shown as stricken text, additions shown as underlined text):

1. ~~For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.~~ For fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values.

In Table 15-16, "Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores with Replacement Steam Generators," of the UFSAR, under the "Fuel Handling Accident for Single Fuel Assembly Event" heading, the licensee proposed to replace the total effective dose equivalent (TEDE) "TEDE at EAB" dose value (in rem) of 1.33 with 1.18, the "TEDE at LPZ" value of 0.14 with 0.13, and the "TEDE in Control Room" value of 2.45 with 2.19. Under the "Fuel Cask Handling Accident for Multiple Fuel Assembly Event" heading, the licensee proposed to replace the "TEDE at EAB" value of 2.05 with 1.93, the "TEDE at LPZ" value of 0.22 with 0.21, and the "TEDE in Control Room" value of 4.05 with 3.62.

2.3 Regulatory Review

The NRC staff considered the following regulatory requirements, licensing and design basis information, and guidance during its review of the proposed changes.

Regulatory Requirements

The NRC staff evaluated the licensee's proposed change to its licensing basis against the radiological dose requirements specified in Section 50.67, "Accident source term."

The principal design criteria for Oconee were developed in consideration of the 70 GDCs for nuclear power plant construction permits proposed by the Atomic Energy Commission in a proposed rulemaking published for 10 CFR Part 50 in the *Federal Register* on July 11, 1967. The Oconee final safety analysis report (as updated) Section 3.1.11 discusses how Oconee meets principal design criteria 11, "Control Room (Category B)."

Design Basis

The Atomic Energy Commission (AEC) issued the construction permits for Oconee on November 6, 1967. The AEC issued the operating licenses for each of the three units on February 6, 1973, October 6, 1973, and July 19, 1974, respectively. The plants' general design criteria (GDC) are discussed in the UFSAR, Chapter 3.1, "Conformance with NRC General Design Criteria," and in the applicable UFSAR sections. The AEC published the final rule that added Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the FR (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992,[1] the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for Oconee are those in the UFSAR. As discussed in the UFSAR, the licensee made changes to the facilities and committed to some of the GDC from 10 CFR Part 50, Appendix A. Based on its review of UFSAR, Section 3.1 and the licensee's submittals, the NRC staff identified the following GDC as being applicable to the proposed amendment.

- UFSAR, Chapter 3, Section 3.1.11, "Criterion 11 – Control Room (Category B)," which states that "Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits."

Guidance

RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," proves guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This regulatory guide endorses a source term derived from NUREG-1465 and provides guidance on the acceptable attributes of other ASTs.

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[1] U.S. Nuclear Regulatory Commission, SECY-92-223 – *Resolution of Deviations Identified During the Systematic Evaluation Program*, September 18, 1992, ADAMS Accession No. ML003763736.

NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000 (ADAMS Accession No. ML003734190).

Technical Documents

NUREG/CR-7022, Volume 2, “FRAPCON-3.4: Integral Assessment,” dated March 2011 (ADAMS Accession No. ML11101A006). An integral assessment was performed for the NRC by Pacific Northwest National Laboratory (PNNL) to quantify the predictive capabilities of FRAPCON-3, a steady-state fuel behavior code designed to analyze fuel behavior from beginning-of-life to rod-average burnup levels of 62 gigawatt-days per metric ton of uranium.

Pacific Northwest National Laboratory (PNNL) report PNNL-18212, Revision 1, “Update of Gap Release Fractions for Non-LOCA [loss-of-coolant-event] Events Utilizing the Revised ANS 5.4 Standard,” dated June 2011 (ADAMS Accession No. ML112070118). This report provides the technical basis for a revision to the non-LOCA fission product gap inventories in NRC RG 1.183.

American Nuclear Society (ANS) 5.4 standard, “Methods for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuels.” The ANS 5.4 standard provides a methodology for determining the radioactive fission product releases for use in assessing radiological consequences of postulated accidents that do not involve abrupt power transients

3.0 TECHNICAL EVALUATION

In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The NRC staff evaluated the licensee’s application to determine if the proposed changes are consistent with the regulations and licensing and design basis information discussed in Section 2 of this safety evaluation. The NRC staff reviewed the acceptability of the proposed changes for conformance to the design bases described in the UFSAR and compliance with 10 CFR 50.67.

This safety evaluation addresses the impact of the proposed changes on previously analyzed design basis accidents radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the NRC staff based its acceptance are the accident radiation dose values in 10 CFR 50.67, and the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

3.1 Fission Gap Analysis

The NRC staff performed a confirmatory FRAPCON-4 (fuel rod performance code) calculation using Oconee’s bounding radial power profile (provided in the licensee’s letter dated November 1, 2018), to provide additional assurance that the release fractions (i.e., radionuclide gap fractions) proposed by the licensee are conservative.

3.1.1 Input and Assumptions

The following inputs and assumptions were used in this calculation.

3.1.1.1 Fuel Rod Design Specifications

The generic Pressurized Water Reactor (PWR) 15x15 fuel rod dimensions, embedded within the FRAPCON input generator spreadsheet were used in this confirmatory calculation. Any differences in fuel rod design specifications relative to the AREVA 15x15 Mark-B-HTP fuel assembly design used in the Oconee reactors are expected to be small and their impact insignificant for this evaluation.

Consistent with the analytical procedures described in Pacific Northwest National Laboratory (PNNL) report PNNL-18212, Revision 1, nominal fuel design specifications (excluding tolerances) were used.

3.1.1.2 Fuel Rod Power History

The Oconee fuel rod radial power profile provided in Table 1, "Projected Rod Powers in the Gap Release Analysis," of the licensee's letter dated November 1, 2018, was used in this calculation.

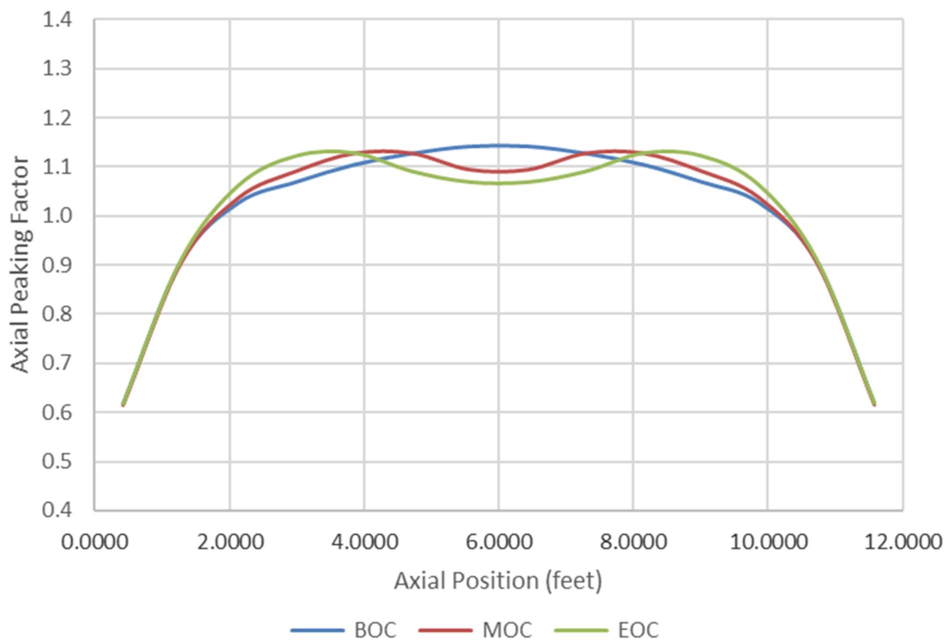
3.1.1.3 Axial Power Distribution

In commercial PWR core reload designs, axial power distribution (AXPD) begins as a cosine shape and then migrates toward a saddle shape at the end of each reload. AXPDs for upcoming Oconee reload cycles are unknown and are expected to vary slightly from cycle-to-cycle. Sensitivity cases were run to investigate the impact of AXPD on accumulated long-lived release fractions. Short-lived release fractions do not accumulate (due to decay) and, therefore, are more sensitive to peaks and knees in the power profile.

At fuel rod exposures approaching the allowable burnup limit of 62 GWD/MTU rod average, peak pellet exposure should be around 67-69 GWD/MTU in a commercial PWR. Hence, both fuel rod radial peaking factor (Fr) and axial peaking factor (Fz) must be considered in defining a bounding power profile. To maximize axial power distribution, axial burnup distribution, axial fuel temperature distribution and the amount of fission gas released along the entire active stack height, a larger radial peaking factor should be employed in combination with a smaller axial peaking factor. These values should be selected to achieve, simultaneously, a rod average burnup of 62 GWD/MTU and a peak pellet burnup of approximately 69 GWD/MTU. The flatter the axial power profile, the larger the number of axial nodes approaching both limits.

The burnup-dependent AXPD selected for this confirmatory calculation is shown in the below Figure 2-1. These shapes are similar to those used to revise RG 1.183, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap" (see Figure A-1 of PNNL-18212, Revision 1). With maximum axial peaking factors (Fz) ranging from 1.143 (beginning-of-cycle (BOC)) to 1.128 (end-of-cycle (EOC)), these axial power distributions are conservatively flat. The combination of these burnup-dependent AXPDs, in combination with the Oconee bounding radial power profile, achieved a peak pellet exposure of 69.3 GWD/MTU at the end-of-life (EOL) burnup target of 62 GWD/MTU.

Figure 2-1: Generic PWR AXPDs



Note: MOC stands for middle-of-cycle.

FRAPCON Output:

```
          xxxxxxxxxxxxxxxxxxxx  input axial shape number   1  xxxxxxxxxxxxxxxxxxxx peak node is  8

 x(   1)=  0.0000  x(   2)=  0.4750  x(   3)=  1.4800  x(   4)=  2.4850  x(   5)=  3.4900  x(   6)=  4.4950  x(   7)=  5.5000
 x(   8)=  6.5050  x(   9)=  7.5100  x(  10)=  8.5150  x(  11)=  9.5200  x(  12)= 10.5250  x(  13)= 11.5300  x(  14)= 12.0000
qf(   1)=  0.5500 qf(   2)=  0.6000 qf(   3)=  1.0000 qf(   4)=  1.0500 qf(   5)=  1.1000 qf(   6)=  1.1300 qf(   7)=  1.1500
qf(   8)=  1.1500 qf(   9)=  1.1300 qf(  10)=  1.1000 qf(  11)=  1.0500 qf(  12)=  1.0000 qf(  13)=  0.6000 qf(  14)=  0.5500

          xxxxxxxxxxxxxxxxxxxx  input axial shape number   2  xxxxxxxxxxxxxxxxxxxx peak node is 10

 x(  15)=  0.0000  x(  16)=  0.4750  x(  17)=  1.4800  x(  18)=  2.4850  x(  19)=  3.4900  x(  20)=  4.4950  x(  21)=  5.5000
 x(  22)=  6.5050  x(  23)=  7.5100  x(  24)=  8.5150  x(  25)=  9.5200  x(  26)= 10.5250  x(  27)= 11.5300  x(  28)= 12.0000
qf(  15)=  0.5500 qf(  16)=  0.6000 qf(  17)=  1.0000 qf(  18)=  1.0700 qf(  19)=  1.1300 qf(  20)=  1.1500 qf(  21)=  1.1000
qf(  22)=  1.1000 qf(  23)=  1.1500 qf(  24)=  1.1300 qf(  25)=  1.0700 qf(  26)=  1.0000 qf(  27)=  0.6000 qf(  28)=  0.5500

          xxxxxxxxxxxxxxxxxxxx  input axial shape number   3  xxxxxxxxxxxxxxxxxxxx peak node is  5

 x(  29)=  0.0000  x(  30)=  0.4750  x(  31)=  1.4800  x(  32)=  2.4850  x(  33)=  3.4900  x(  34)=  4.4950  x(  35)=  5.5000
 x(  36)=  6.5050  x(  37)=  7.5100  x(  38)=  8.5150  x(  39)=  9.5200  x(  40)= 10.5250  x(  41)= 11.5300  x(  42)= 12.0000
qf(  29)=  0.5500 qf(  30)=  0.6000 qf(  31)=  1.0000 qf(  32)=  1.1000 qf(  33)=  1.1500 qf(  34)=  1.1000 qf(  35)=  1.0700
qf(  36)=  1.0700 qf(  37)=  1.1000 qf(  38)=  1.1500 qf(  39)=  1.1000 qf(  40)=  1.0000 qf(  41)=  0.6000 qf(  42)=  0.5500

   increment      axial station           normalized
                 feet         meters     heat flux
                                            1st        2nd        3rd        4th        5th        6th        7th        8th
      1        0.4286      0.13063       0.6169     0.6149     0.6180
      2        1.2857      0.39189       0.9069     0.9046     0.9100
      3        2.1429      0.65314       1.0278     1.0374     1.0621
      4        3.0000      0.91440       1.0702     1.0916     1.1218
      5        3.8571      1.17566       1.1053     1.1277     1.1277
      6        4.7143      1.43691       1.1284     1.1278     1.0903
      7        5.5714      1.69817       1.1427     1.0945     1.0686
      8        6.4286      1.95943       1.1428     1.0944     1.0685
      9        7.2857      2.22069       1.1285     1.1277     1.0901
     10        8.1429      2.48194       1.1055     1.1279     1.1275
     11        9.0000      2.74320       1.0705     1.0919     1.1221
     12        9.8571      3.00446       1.0280     1.0377     1.0626
     13       10.7143      3.26571       0.9084     0.9061     0.9116
     14       11.5714      3.52697       0.6180     0.6159     0.6190
```

3.1.2 FRAPCON-4 Confirmatory Calculations

3.1.2.1 AXPD Sensitivity Cases

AXPDs for upcoming Oconee reload cycles are unknown and are expected to vary slightly from cycle-to-cycle. Furthermore, fuel assemblies may achieve discharge burnup in two or three reload cycles, depending on operating power history. Using the conservative generic AXPD shown in Figure 2.1, FRAPCON-4 cases were run to investigate the impact of different operating power histories on predicted long-term nuclide release fractions.

Table 3-1, below, provides the results of this sensitivity study. Examination of this table reveals that the calculated release fractions are relatively insensitive to the variation in sequencing of the selected PWR AXPDs. So, whether the EOL exposure was reached in one unrealistic cycle, two long cycles, or a more representative three cycles of operation, the results are essentially unchanged.

Table 3-1: AXPD Sensitivity Study

<table>
<tr><th>Rod Power (Fr)</th><th>AXPD (Fz)</th><th>Fission Gas Release (FGR) Uncertainty</th><th>Calculated Release Fraction (Kr-85)</th></tr>
<tr><td rowspan="2">Bounding
Fr = 1.65, 0 – 50 BU

Linear decrease
Fr = 1.25 at 62 BU</td><td>Case 3x1
3 AXPDs for lifetime</td><td>Nominal</td><td>0.240

(69.3 pellet BU)</td></tr>
<tr><td>Case 3x3
3 AXPDs for 3 cycles</td><td>Nominal</td><td>0.241

(69.3 pellet BU)</td></tr>
</table>

3.1.2.2 Long-Lived Nuclide Release Fractions - Application of Fission Gas Release Modelling Uncertainty

As shown above, FRAPCON-4 calculated a long-term release fraction of 0.241 based on the Oconee radial power profile and a conservative burn-up dependent AXPD. However, FRAPCON-4 is a best-estimate code. Appendix C of PNNL-18212, Revision 1, provides an acceptable analytical technique for calculating gap release fractions including the application of modelling uncertainties. This analytical procedure is being issued within an upcoming revision to RG 1.183 (currently within Draft Regulatory Guide (DG)-1199 and DG-1327). Following this procedure, 95/95 gap release fractions were calculated. As described in NUREG/CR-7022, Volume 2, the standard deviation for FRAPCON's steady-state FGR prediction is 2.6% absolute, up to 70 GWD/MTU. Based on the extent of FRAPCON's validation database, a 2.36-sigma (σ) level of confidence is recommended (from PNNL-18212, Revision 1).

Krypton (Kr)-$85_{95/95}$ $= [\ (\text{EOL FGR})_{\text{FRAPCON-4}} + (k\sigma)_{\text{FRAPCON-4}}\ ]$

$= [\ 0.241 + (2.36*0.026)\ ]$

$= 0.302$

Cesium (Cs)-$134{,}137_{95/95}$ $= [\ ((\text{EOL FGR})_{\text{FRAPCON-4}} * (2.0)^{0.5}) + (k\sigma)_{\text{FRAPCON-4}}\ ]$

$= [\ (0.241*2^{0.5}) + (2.36*0.026)\ ]$

$= 0.402$

Since the development of the analytical guidance in PNNL-18212, Revision 1, FRAPCON has been revised to update several important models and introduce the ability to apply pre-determined modelling uncertainties. PNNL-18212, Revision 1, employed FRAPCON-3.3; whereas the current version used in this assessment is FRAPCON-4. Taking advantage of these new features, a sensitivity study was completed to investigate the impact of FGR model uncertainties on predicted long-lived nuclide gap fractions.

Table 3-2: FGR Modelling Uncertainties

Rod Power (Fr)	AXPD (Fz)	FGR Uncertainty	Calculated Release Fraction (Kr-85)
Bounding Fr = 1.65, 0 – 50 BU Linear decrease Fr = 1.25 at 62 BU	Case 3x3 3 AXPDs for lifetime	Nominal	0.241
		1σ	0.281
		1.5σ	0.296
		2σ	0.314

Examination of above Table 3-2 reveals, as expected, progressive increases in predicted Kr-85 release with application of higher FGR model uncertainties. Unlike the additive uncertainty applied to the final FGR prediction, the compounding effect of applying an embedded uncertainty factor to the FGR model at each burnup step promotes higher fuel temperatures and slightly more accumulated FGR at EOL. For example, the applied 2σ case predicted a release fraction of 0.314 compared with a release fraction of 0.293 for an additive 2σ case (0.241 x 2(0.026) = 0.293).

The derivation of Cs-134,137 release fractions changes with the applied FGR model uncertainty, relative to the earlier additive FGR uncertainty. Multiplying the Kr-85 release fraction by $(2.0)^{0.5}$, to account for a factor of 2.0 higher Cs diffusion coefficient, produces the following upper tolerance release fractions.

Nuclide	Additive Uncertainty	Applied Model Uncertainty
Cs-134,137	0.402	0.397 (1σ)
		0.419 (1.5σ)
		0.444 (2σ)

3.1.3 Comparison of FRAPCON-4 and COPERNIC Long-Lived Release Fractions

Table 4, “Results from Gap Release Calculations,” of the licensee’s letter dated November 1, 2018, provides the 95/95 gap release fractions calculated using the COPERNIC fuel rod thermal-mechanical code, bounding radial power distribution, and Oconee-specific design parameters and AXPDs. As shown in Table 3-3 below, a comparison of the predicted 95/95 gap release fractions for the long-lived nuclides reveals good agreement. In all cases, COPERNIC calculated values are slightly more conservative.

Table 3-3: Comparison of COPERIC and FRAPCON-4 Release Fractions

Nuclide	COPERNIC (Duke)	FRAPCON-4	
		Additive Uncertainty	Applied 2σ FGR Model Uncertainty
Kr-85	0.318	0.302	0.314
Cs-134,137	0.450	0.402	0.444

Table 5, "Bounding Gap Fractions for Applications on ONS Fuel Handling Accidents," of the licensee's letter dated November 1, 2018, provides the bounding release fractions which are applied in the Oconee fuel handling accident (FHA). The licensee opted to apply a factor of 4x on the original RG 1.183, Table 3 values for the long-lived nuclides. Specifically, employing gap release fractions of 0.40 for Kr-85 and 0.48 for Cs-134,137. This approach adds more conservatism relative to the calculated values above.

3.1.4 Comparison of FRAPCON-4 and COPERNIC Short-Lived Release Fractions

FRAPCON-4 has the capability of calculating release fractions (actually release/birth ratios (R/B)) for many short-lived nuclides based on the 2011 ANS-5.4 standard. The limiting FRAPCON-4 case for the long-lived release fractions was re-run with the 2011 ANS-5.4 standard turned on. No embedded FGR model uncertainty was applied. As expected, the peak R/B occurred at the knee in the power profile (50 GWD/MTU). In accordance with the ANS-5.4 standard, the best-estimate FRAPCON-4 predictions were multiplied by a 5.0 uncertainty factor. Table 4 of the licensee's letter dated November 1, 2018, provides the 95/95 gap release fractions calculated using the COPERNIC fuel rod thermal-mechanical code, bounding radial power distribution, and Oconee specific design parameters and AXPDs. Table 3-4, below, provides a comparison of the COPERNIC and FRAPCON-4 calculated short-lived nuclide R/Bs. The FRAPCON-4 predictions are slightly larger with a maximum difference of 0.0044. This difference is reasonable, especially since these upper tolerance values include a 5.0 uncertainty multiplier.

Table 3-4, below, also compares the calculated gap fractions to those employed in the Oconee FHA dose assessment. The licensee opted to maintain the existing RG 1.183 Table 3 values for the short-lived nuclides; even though the COPERNIC calculations using the 2011 ANS-5.4 standard showed significant margins. The FRAPCON-4 calculations confirm the overall conservatism of the proposed short-lived gap fractions.

Table 3-4: Comparison of COPERIC and FRAPCON-4 Release Fractions

Nuclide	COPERNIC (the licensee)	FRAPCON-4	Oconee Gap Fraction
Kr-85m	0.0066	0.0083	0.05
Kr-87	0.0034	0.0043	0.05
Kr-88	0.0047	0.0059	0.05
Xenon (Xe)-133	0.0150	0.0188	0.05
Xe-135	0.0086	0.0107	0.05
Iodine (I)-131	0.0158	0.0198	0.08
I-132	0.0179	0.0223	0.05
I-133	0.0095	0.0119	0.05
I-135	0.0068	0.0085	0.05

Utilizing the analytical procedure for calculating high confidence radionuclide release fraction in PNNL-18212, Revision 1, the NRC staff completed independent confirmatory FRAPCON-4 calculations. The results of these calculations provide additional assurance that the Oconee release fractions (i.e., radionuclide gap fractions) proposed by the licensee are conservative.

3.2 Fuel Handling and Fuel Cask Handling Accident Analyses

The licensee proposes gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft LHGR limit in Footnote 11 of Table 3 in RG 1.183. The non-LOCA gap fractions stated in Table 3 of RG 1.183 are applied to the non-LOCA accidents if fuel failure occurs during the accident. The following accidents at Oconee assume fuel failure: FHA, locked rotor accident (LRA), control rod ejection accident (CREA) and fuel cask handling accident. The licensee states that no non-LOCA accidents that may result in departure from nucleate boiling are considered (e.g., LRA, CREA) because the fuel cycles for Oconee will be designed so that no fuel rod predicted to enter departure from nucleate boiling will have been operated beyond the current limit in RG 1.183, Footnote 11 for maximum LHGR. However, the license amendment request did not incorporate this new design requirement into the licensing basis. Therefore, the NRC staff issued a request for additional information (RAI) on February 2, 2019 (ADAMS Accession No. ML19053A562), requesting that the licensee describe how it planned to incorporate the new design requirement into the Oconee licensing basis as reflected in the updated final safety analysis report, TSs, or any other document controlled under 10 CFR 50.59 (such as the core operating limits report).

In its response to the RAI, dated March 7, 2019, the licensee stated that they will incorporate the new design requirement into the Oconee licensing basis via an attached proposed revision to the UFSAR, Section 15.1.10, which supersedes the UFSAR Section 15.1.10 mark-up included in the November 1, 2018 license amendment request. The remainder of the originally proposed UFSAR changes were not affected by the licensee's response to the NRC staff's request. The proposed revision to UFSAR Section 15.1.10 regarding departure from nucleate boiling (DNB) states:

> For non-DNB fuel pins that exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values … The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident. The fuel cycle design also ensures that no fuel rod predicted to experience DNB in any other non-LOCA accidents (e.g., locked rotor accident or rod ejection accident) will have operated beyond the power/burnup criteria of Footnote 11 in Regulatory Guide 1.183 and that the gap fractions used in these non-LOCA accident analyses remain those stated in Table 3 of RG 1.183.

The licensee proposes to apply the following proposed bounding gap fractions to the Oconee FHA:

Isotope or isotope group	Gap fractions from Table 3 of RG 1.183, Revision 0	Current licensing basis gap fractions	Proposed bounding gap fractions
I-131	0.08	0.16	0.08
Kr-85	0.10	0.30	0.40
Other Noble Gases	0.05	0.10	0.05
Other Halogens	0.05	0.10	0.05
Cs-134 (Alkali Metal)	0.12	0.36	0.48
Cs-137 (Alkali Metal)	0.12	0.36	0.48
Other Alkali Metals	0.12	0.24	0.12

3.2.1 Fuel Handling Accident

The licensee considered two fuel handling events for the postulated FHA: (1) the drop of a single fuel assembly in the containment and in the spent fuel pool, and (2) the drop of a fuel transport cask or an independent spent fuel storage installation (ISFSI) transfer cask (with multiple fuel assemblies) in the spent fuel pool. The FHA involves dropping a single fuel assembly during fuel handling operations, causing mechanical damage with a subsequent release of fission products. The licensee concluded that the radiological consequences resulting from the postulated FHA are within the dose acceptance criteria specified in RG 1.183 and in 10 CFR 50.67 for the exclusion area boundary, low population zone, and control room operator.

The licensee reached this conclusion as a result of: (1) implementing the gap fractions in the above table, (2) assuming the source term is unchanged from the current licensing basis source term, and (3) assuming all other inputs and assumptions are unchanged from the postulated FHA from the current licensing basis, as reflected in the UFSAR Chapter 15.

In the current licensing basis, for multiple fuel assembly events involving a drop of a fuel transport cask or an ISFSI transfer cask in the spent fuel pool, the number of fuel assemblies involved in these events varies depending on the location of the cask drop (i.e., the Unit 1 and 2, or Unit 3 spent fuel pool) and the type of cask dropped (i.e., transport cask or ISFSI cask). Oconee is designed with two spent fuel pools, the Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. The number of fuel assemblies that would be involved in each cask drop event in each spent fuel pool ranges from 518 to 1024 fuel assemblies and the number of fuel assemblies involved in these events is greater than the amount of fuel recently discharged from a core (i.e., 177 fuel assemblies per core).

The licensee assumes two different decay times for each event: one for the fuel recently discharged from the core (55 to 70 days) and one for the other fuel involved in the event (1 year). The licensee performed 12 different radiological consequence cases for these FHA events, using a combination of different type of cask dropped, different location of the cask drop, and different control room intakes (i.e., Unit 1 and 2, or Unit 3, control room). The licensee determined that (1) the bounding single fuel assembly FHA event is the FHA in either the Unit 1 and 2, or Unit 3, spent fuel pool to the Unit 1 and 2 control room air intake, and (2) the bounding

multiple-fuel assembly FHA event is the transport cask drop event in either the Unit 1 and 2, or Unit 3, spent fuel pool to the Unit 1 and 2 control room air intake.

The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the FHA events that produce the greatest radiological consequence. The radiological consequences calculated by the NRC staff are within the dose acceptance criteria specified in 10 CFR 50.67 and meet the dose acceptance criteria specified in RG 1.183. Even though the NRC staff performed its confirmatory dose calculations, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are summarized in the below.

Radiological Dose Results (total effective dose equivalent in rem)			
Accident	Current FSAR	Licensee Dose Results	SRP 15.0.1 and RG 1.183 Dose Acceptance Criteria
Fuel Handling Accident (single assembly event)			
Exclusion Area Boundary	1.33	1.18	6.3
Low Population Zone	0.14	0.13	6.3
Control Room	2.45	2.19	5.0
Fuel Handling Accident (multiple assembly event)			
Exclusion Area Boundary	2.05	1.93	6.3
Low Population Zone	0.22	0.21	6.3
Control Room	4.05	3.62	5.0

The radiological consequences at the exclusion area boundary, low population zone, and in the control room calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and meet the accident-specific dose acceptance criteria specified in RG 1.183 and SRP 15.0.1.

3.3 Technical Evaluation Summary

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the exclusion area boundary, low population zone, and control room are within the radiological dose guidelines provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1 and RG 1.183. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 above. The licensee's calculated dose results are given in Table 1. To verify the licensee's analyses, the NRC staff performed confirmatory radiological consequence dose calculations. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations.

Based on its evaluation as documented in Sections 3.1 and 3.2 of this safety evaluation, the NRC staff concludes that the licensee's proposed changes are acceptable. The radiological consequences calculated by the NRC staff for the exclusion area boundary, low population zone, and control room are consistent with those estimated by the licensee. Moreover, the radiological consequences calculated by both the licensee and the NRC staff are well within the

radiation dose limits set forth in 10 CFR 50.67 and principal design criteria 11, and within the accident-specific criteria stated in SRP Section 15.0.1 and RG 1.183. The NRC staff finds reasonable assurance that the exclusion area boundary, low population zone, and control room doses estimated by the licensee for the FHA will comply with the applicable accident dose criteria and are acceptable; therefore, the licensee's proposed revision to the gap release fractions for high-burnup fuel rods (i.e., peak burnup up to 62 GWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding than 54 GWD/MTU, as detailed in Table 3 of RG 1.183) is acceptable. The NRC staff concludes that proposed changes meet the requirements of 10 CFR 50.67 and principal design criteria 11, and are within the accident-specific criteria stated in SRP Section 15.0.1 and RG 1.183, and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of South Carolina official of the proposed issuance of the amendments on June 17, 2019 (ADAMS Accession No. ML19170A268). The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement 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 has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding, which was published in the *Federal Register* on January 31, 2019 (84 FR 811). 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

Based on the aforementioned considerations, the Commission has concluded 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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