



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

October 13, 2020

Ms. Kim Maza
Site Vice President
Shearon Harris Nuclear Power Plant
Mail Code NHP01
5413 Shearon Harris Road
New Hill, NC 27562-9300

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - CORRECTION
LETTER TO AMENDMENT NO. 179 REGARDING DEPARTURE FROM
NUCLEATE BOILING RATIO SAFETY LIMIT TO ADDRESS TRANSITION TO
NEW FUEL TYPE (EPID L-2019-LLA-0076)

Dear Ms. Maza:

On September 29, 2020, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 179 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant (Harris), Unit 1. The amendment revised Technical Specification (TS) requirements in response to your application dated April 10, 2019, as supplemented by letters dated June 6, 2019 and December 20, 2019.

Subsequent to issuance of the amendment, it was determined that an incorrect amendment number was used for Shearon Harris Nuclear Power Plant, Unit 1. The September 29, 2020, letter and enclosures listed Amendment No. 179 for Renewed Facility Operating License No. NPF-63. Amendment No. 180 is the number that should have been used.

Enclosed with this letter are corrected pages that we ask you to replace for the previously issued September 29, 2020, amendment. Change bars are added where corrected numbers appear in the enclosed document. The NRC concludes that the correction is entirely editorial in nature and does not change the staff's previous conclusion in the safety evaluation for Amendment No. 179, nor does it affect the no significant hazards consideration, as published in the *Federal Register* on March 30, 2020 (85 FR 17601). The corrected Renewed Facility Operating License page 4, amendment pages, safety evaluation page, and TS pages 2-1, 6-24a, 6-24b, 6-24c, 6-24d are enclosed.

If you have any questions regarding this matter, please contact me at (301) 415-1387 or by e-mail at Tanya.Hood@nrc.gov.

Sincerely,

/RA/

Tanya E. Hood, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:

1. Corrected amendment pages
2. Corrected safety evaluation and license pages

cc: Listserv

Enclosure 1

Corrected amendment pages



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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 180
Renewed License No. NPF-63

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated April 10, 2019, as supplemented by letters dated June 6, 2019 and December 20, 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-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 180, are hereby incorporated into this license. Duke Energy Progress, LLC 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 the startup of Cycle 24.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine S. Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: September 29, 2020

Enclosure 2

Corrected safety evaluation and license pages

ATTACHMENT TO LICENSE AMENDMENT NO. 180
SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
RENEWED FACILITY OPERATING LICENSE NO. NPF-63
DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change:

Remove
Page 4

Insert
Page 4

Replace the following pages of the 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

2-1
6-24a
6-24b
6-24c
6-24d

2-1
6-24a
6-24b
6-24c
6-24d

- C. This 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

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 180, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC, shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)¹

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

¹The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

* On April 29, 2013, the name of "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

- 2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:
- The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation for HTP fuel and ≥ 1.12 for the ORFEO-GAIA DNB correlation for GAIA fuel.
 - The peak centerline temperature shall be maintained $< [4901 - (1.37 \times 10^{-3} \times (\text{Burnup, MWD/MTU}))]$ °F.

APPLICABILITY: MODES 1 and 2.

ACTION:

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

- 2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig except during hydrostatic testing.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

- 2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).
- d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).
- e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.
(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).
- f. ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.
(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.
(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).
- h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.
(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).
- i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).
- j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- k. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods."
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.2 – SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 – DNB Parameters, and 3.9.1 – Boron Concentration).
- l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.
(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).
- m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.
(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.
(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).
- o. Mechanical Design Methodologies
XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.
ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.
XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.
ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.
XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.
EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.
BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- (Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).
- p. DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," approved version as specified in the COLR.
(Methodology for Specification 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)
- q. DPC-NE-1008-P-A, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," as approved by NRC Safety Evaluation dated May 18, 2017.
(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 – Boron Concentration).
- r. DPC-NF-2010-A, "Nuclear Physics Methodology for Reload Design," as approved by NRC Safety Evaluation dated May 18, 2017.
(Methodology for Specifications 3.1.1.1 – SHUTDOWN MARGIN – MODES 1 and 2, 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.2.5 – Borated Water Source – Shutdown, 3.1.2.6 – Borated Water Sources – Operating, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.5.1 – ECCS Accumulators – Cold Leg Injection, 3.5.4 – ECCS Refueling Water Storage Tank, and 3.9.1 – Boron Concentration).
- s. DPC-NE-2011-P-A, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated May 18, 2017.
(Methodology for Specification 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).
- t. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018.
(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).
- u. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018.
(Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

- v. ANP-10341P-A, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations," approved version as specified in the COLR.
(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.I. The report shall include:

- a. The scope of inspections performed on each SG,
- b. Degradation mechanisms found,
- c. Nondestructive examination techniques utilized for each degradation mechanism,
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
- e. Number of tubes plugged during the inspection outage for each degradation mechanism,
- f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated April 10, 2019¹, as supplemented by letters dated June 6, 2019² and December 20, 2019³, Duke Energy Progress, LLC (the licensee) submitted a license amendment request (LAR) for changes to the Shearon Harris Nuclear Power Plant, Unit 1 (Harris), Technical Specifications (TSs). The requested changes would modify TS 2.1.1.a to add the departure from nucleate boiling ratio (DNBR) safety limit associated with the transition from the high thermal performance fuel to a fuel assembly design with characteristics similar to the GAIA fuel design using the ORFEO-GAIA correlation methodology. In addition, TS 6.9.1.6.2 would be revised to include the U.S. Nuclear Regulatory Commission (NRC or the Commission) final safety evaluation report (SER)⁴ approving topical report, ANP-10341-P-A, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations," Revision 0,⁵ as a Core Operating Limits Report (COLR) reference for the NRC-approved departure from nucleate boiling correlation associated with the GAIA fuel design for applicable fuel specifications and characteristics. The GAIA fuel design is limited to the parameters outlined in Attachment 7 to the LAR in Appendix J, "Harris Plant Specific Data for GAIA Fuel Application of the ORFEO-GAIA Correlation to the GAIA Fuel Design," to DPC-NE-2005-P, "Thermal-Hydraulic Statistical Core Design Methodology," Revision 6. This safety evaluation does not approve the use of GAIA fuel nor does it approve the use of the generically approved GAIA topical report in ANP-10342-P-A, "GAIA Fuel Assembly Mechanical Design."⁶

The amendment would add Appendix J to DPC-NE-2005-P which includes Harris plant specific data of fuel specifications and characteristics related to GAIA fuel using the ORFEO-GAIA critical heat flux correlation. Revision 6 discussed the methodology to address the applicability of the ORFEO-GAIA critical heat flux use at Harris to perform the statistical core design analysis in-house when applying the ORFEO-GAIA critical heat flux correlation at Harris. The TSs would

¹ Agencywide Documents Access and Management System (ADAMS) Accession No. ML19100A442

² ADAMS Accession No. ML19157A036

³ ADAMS Accession No. ML19354B380

⁴ ADAMS Accession No. ML18236A371

⁵ ADAMS Accession No. ML16238A078

⁶ ADAMS Accession No. ML19309D913

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - CORRECTION
LETTER TO AMENDMENT NO. 179 REGARDING DEPARTURE FROM
NUCLEATE BOILING RATIO SAFETY LIMIT TO ADDRESS TRANSITION TO
NEW FUEL TYPE (EPID L-2019-LLA-0076) DATED OCTOBER 13, 2020

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