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

August 15, 2018

Mr. Steven Capps
Senior Vice President
Nuclear Corporate
Duke Energy Corporation
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2; MCGUIRE NUCLEAR STATION, UNITS 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENTS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (EPID L-2017-LLA-0377)

Dear Mr. Capps:

The U.S. Nuclear Regulatory Commission (NRC) has issued the following enclosed amendments: Amendment Nos. 299 and 295 to Renewed Facility Operating License Nos. NPF-35 and NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively; Amendment Nos. 309 and 288 to Renewed Facility Operating License Nos. NPF-9 and NPF-17 for the McGuire Nuclear Station, Units 1 and 2, respectively; Amendment Nos. 409, 411, and 410 to Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, respectively; Amendment No. 166 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1; and Amendment No. 259 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2.

The amendments revise the technical specifications (TSs) in response to the Duke Energy Carolinas, LLC and Duke Energy Progress, LLC application dated November 7, 2017. The amendments revise the TSs for each of these facilities based on Technical Specifications Task Force (TSTF) Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing."

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

If you have any questions, please contact me at (301) 415-6256 or by e-mail at Dennis.Galvin@nrc.gov.

Sincerely,

Dennis J. Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369, 50-370, 50-269, 50-270, 50-287, 50-400, and 50-261

Enclosures:

1. Amendment No. 299 to NPF-35
2. Amendment No. 295 to NPF-52
3. Amendment No. 309 to NPF-9
4. Amendment No. 288 to NPF-17
5. Amendment No. 409 to DPR-38
6. Amendment No. 411 to DPR-47
7. Amendment No. 410 to DPR-55
8. Amendment No. 166 to NPF-63
9. Amendment No. 259 to DPR-23
10. Safety Evaluation

cc: Mr. Robert T. Simril
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

Mr. Thomas Ray
Site Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

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

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

Ms. Tanya Hamilton
Site Vice President
Duke Energy Progress, LLC
Shearon Harris Nuclear Power Plant, Unit 1
5413 Shearon Harris Road, M/C HNP01
New Hill, NC 27562-0165

Listserv

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 299
Renewed License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Carolinas, LLC, acting for itself, and North Carolina Electric Membership Corporation (licensees), dated November 7, 2017, 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, 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-35 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-35
and Technical Specifications

Date of Issuance: August 15, 2018

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated November 7, 2017, 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, 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-52 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 295, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-52
and the Technical Specifications

Date of Issuance: August 15, 2018

ATTACHMENT TO

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

LICENSE AMENDMENT NO. 299

RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 295

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) 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	License
NPF-35, page 4	NPF-35, page 4
NPF-52, page 4	NPF-52, page 4
TSs	TSs
1.1-3	1.1-3
3.4.10-2	3.4.10-2
3.4.14-3	3.4.14-3
3.5.2-2	3.5.2-2
3.6.3-6	3.6.3-6
3.6.6-2	3.6.6-2
3.7.1-2	3.7.1-2
3.7.2-2	3.7.2-2
3.7.3-2	3.7.3-2
3.7.5-3	3.7.5-3
5.5-6	5.5-6

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71 (e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

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

(5) Fire Protection Program

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program that complies with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated September 25, 2013; as supplemented by letters dated January 13, 2015; January 28, 2015; February 27, 2015; March 30, 2015; April 28, 2015; July 15, 2015; August 14, 2015; September 3, 2015; December 11, 2015; January 7, 2016; March 23, 2016; June 15, 2016; August 2, 2016; September 7, 2016; and, January 26, 2017, as approved in the SE dated February 8, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 295, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71 (e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

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

(5) Fire Protection Program

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program that complies with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated September 25, 2013; as supplemented by letters dated January 13, 2015; January 28, 2015; February 27, 2015; March 30, 2015; April 28, 2015; July 15, 2015; August 14, 2015; September 3, 2015; December 11, 2015; January 7, 2016; March 23, 2016; June 15, 2016; August 2, 2016; September 7, 2016; and, January 26, 2017, as approved in the SE dated February 8, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

1.1 Definitions (continued)

DOSE EQUIVALENT Xe-133	DOSE EQUIVALENT Xe-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT Xe-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be $\geq$ 2460 psig and $\leq$ 2510 psig.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	------NOTES------ 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ------------ Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the INSERVICE TESTING PROGRAM, and in accordance with the Surveillance Frequency Control Program AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Number / Position / Function NI162A / Open / SI Cold Leg Injection NI121A / Closed / SI Hot Leg Injection NI152B / Closed / SI Hot Leg Injection NI183B / Closed / RHR Hot Leg Injection NI173A / Open / RHR Cold Leg Injection NI178B / Open / RHR Cold Leg Injection NI100B / Open / SI Pump Suction from RWST NI147B / Open / SI Pump Mini-Flow	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.2	--------------------NOTE-------------------- Not required to be met for system vent flow paths opened under administrative control. -- Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.3	Verify ECCS locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	-----------------------------NOTE---------------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.6	Perform leakage rate testing for Containment Purge System, Hydrogen Purge System, and Containment Air Release and Addition System valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.3	Deleted.	
SR 3.6.6.4	Deleted.	
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to manually start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to manually open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage
SR 3.6.6.8	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	---------------------------------NOTE------------------------------- Only required to be performed prior to entry into MODE 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within ±1%.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	---------------------------------NOTE------------------------------- Only required to be performed prior to entry into MODE 2. --- Verify closure time of each MSIV is within limits on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more MFIV or MFCV bypass valves inoperable.	C.1 Close or isolate bypass valve. AND C.2 Verify bypass valve is closed or isolated.	72 hours Once per 7 days
D. Two valves in the same flow path or the tempering valve inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV, MFCV, their associated bypass valve, and the tempering valve is within limits on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	-----------------------------------NOTE--------------------------- Not applicable to automatic valves when THERMAL POWER is ≤ 10% RTP. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-----------------------------------NOTE--------------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	-----------------------------------NOTE--------------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program (Deleted)

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the

(continued)

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 309
Renewed License No. NPF-9

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

    A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9, filed by the Duke Energy Carolinas, LLC (licensee), dated November 7, 2017, 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.

Enclosure 3

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-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-9 and the Technical Specifications

Date of Issuance: August 15, 2018

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 288
Renewed License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. NPF-17, filed by the Duke Energy Carolinas, LLC (the licensee), dated November 7, 2017, 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, 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-17 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 288 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-17
and the Technical Specifications

Date of Issuance: August 15, 2018

ATTACHMENT TO

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

LICENSE AMENDMENT NO. 309

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 288

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) 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	License
NPF-9, page 3	NPF-9, page 3
NPF-17, page 3	NPF-17, page 3
TSs	TSs
1.1-3	1.1-3
3.4.10-2	3.4.10-2
3.4.14-3	3.4.14-3
3.5.2-3	3.5.2-3
3.6.3-6	3.6.3-6
3.6.6-2	3.6.6-2
3.7.1-2	3.7.1-2
3.7.2-2	3.7.2-2
3.7.3-2	3.7.3-2
3.7.5-3	3.7.5-3
5.5-6	5.5-6

(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;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2, and;

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

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

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

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3469 megawatts thermal (100%).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Updated Final Safety Analysis Report</u>

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than June 12, 2021, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(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;

(5) Pursuant to the Act and 10 CFR Parts, 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as my be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such by product material as may be produced by the Duke Training and Technology Center.

C. This renewed operating 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 thereafter 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 a reactor core full steady state power level of 3469 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 288 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than March 3, 2023, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59, and otherwise complies with the requirements in that section.

1.1 Definitions (continued)

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

   3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

   All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

   LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be ≥ 2460 psig and ≤ 2510 psig.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-----------------------------NOTE----------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. --- Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	In accordance with the INSERVICE TESTING PROGRAM, and in accordance with the Surveillance Frequency Control Program <u>AND</u> Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action or flow through the valve

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each position stop is in the correct position. Centrifugal Charging Pump Injection Throttle Valve Number: NI480, NI481, NI482, NI483 Safety Injection Pump Throttle Valve Number: NI488, NI489, NI490, NI491	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.8	Verify, by visual inspection, that the ECCS containment sump strainer assembly and the associated enclosure are not restricted by debris and show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ----- Verify each containment isolation manual valve and blind flange that is located inside containment or annulus and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.6	Perform leakage rate testing for containment purge lower and upper compartment and incore Instrument room valves with resilient seals.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.3	Not Used	Not Used
SR 3.6.6.4	Not Used	Not Used
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to manually start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to manually open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage
SR 3.6.6.8	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met. **OR** One or more steam generators with less than two MSSVs OPERABLE.	B.1 Be in MODE 3. **AND** B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------------------NOTE-------------------------------- Only required to be performed prior to entry into MODE 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within $\pm 1\%$.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------------------NOTE-------------------------------- Only required to be performed prior to entry into MODE 2. --- Verify closure time of each MSIV is ≤ 8.0 seconds on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more MFCV's bypass valves or MFW/AFW NBVs inoperable.	C.1 Close or isolate MFCV's bypass valve or MFW/AFW NBV. AND C.2 Verify MFCV's bypass valve or MFW/AFW NBV is closed or isolated.	72 hours Once per 7 days
D. Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV, MFCV, MFCV's bypass valve, and MFW/AFW NBV is ≤ 10 seconds on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	-------------------------------NOTE------------------------------- Not applicable to automatic valves when THERMAL POWER is ≤ 10% RTP. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-------------------------------NOTE------------------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	-------------------------------NOTE------------------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

5.5.8 Inservice Testing Program (Deleted)

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

(continued)

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 409
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 No. 1 (the facility), Renewed Facility Operating License No. DPR-38, filed by the Duke Energy Carolinas, LLC (the licensee), dated November 7, 2017, 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, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 409 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Boome

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38
and the Technical Specifications

Date of Issuance: August 15, 2018

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 411
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 No. 2 (the facility), Renewed Facility Operating License No. DPR-47, filed by the Duke Energy Carolinas, LLC (the licensee), dated November 7, 2017, 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, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 411 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47
and the Technical Specifications

Date of Issuance: August 15, 2018

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 410
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 No. 3 (the facility), Renewed Facility Operating License No. DPR-55, filed by the Duke Energy Carolinas, LLC (the licensee), dated November 7, 2017, 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, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 410 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-55 and the Technical Specifications

Date of Issuance: August 15, 2018

ATTACHMENT TO

OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3

LICENSE AMENDMENT NO. 409

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

LICENSE AMENDMENT NO. 411

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

LICENSE AMENDMENT NO. 410

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove	Insert
DPR-38, page 3	DPR-38, page 3
DPR-47, page 3	DPR-47, page 3
DPR-55, page 3	DPR-55, page 3

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	Remove	Insert
1.1-3	1.1-3	3.7.5-4	3.7.5-4
3.4.10-2	3.4.10-2	3.7.10-2	3.7.10-2
3.5.2-5	3.5.2-5	3.7.10-3	3.7.10-3
3.5.3-3	3.5.3-3	3.7.19-1	3.7.19-1
3.6.3-5	3.6.3-5	3.7.19-3	3.7.19-3
3.6.5-4	3.6.5-4	3.10.1-5	3.10.1-5
3.7.1-1	3.7.1-1	5.0-12	5.0-12
3.7.3-2	3.7.3-2	5.0-13	5.0-13

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 409 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 411 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 410 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

1.1 Definitions (continued)

CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of the Environmental Protection Agency (EPA) Federal Guidance Report No. 11.
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	Verify each HPI pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.4	Verify each HPI automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.5	Verify each HPI pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.6	Verify, by visual inspection, each HPI train reactor building sump suction inlet is not restricted by debris and suction inlet strainers show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program
SR 3.5.2.7	Cycle each HPI discharge crossover valve and LPI-HPI flow path discharge valve.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.2	Verify LPI locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.3.3	Verify each LPI pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.3.4	Verify each LPI automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.3.5	Verify each LPI pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.3.6	Verify, by visual inspection, each LPI train reactor building sump suction inlet is not restricted by debris and suction inlet strainers show no evidence of structural distress or abnormal corrosion.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	--------------------------NOTE---------------------- Not required to be met for reactor building spray system vent flow paths opened under administrative control. -- Verify each reactor building spray and cooling manual and non-automatic power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.5.3	Verify each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.5.4	Verify that the containment heat removal capability is sufficient to maintain post accident conditions within design limits.	In accordance with the Surveillance Frequency Control Program

(continued)

3.7 PLANT SYSTEMS

3.7.1 Main Steam Relief Valves (MSRVs)

LCO 3.7.1 Eight MSRVs shall be OPERABLE on each main steam line.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSRVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	------------------------NOTE-------------------------- Only required to be performed in MODES 1 and 2. --- Verify each MSRV lift setpoint in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------------NOTE------------------------- Only required to be performed in MODES 1 and 2. --- Verify the closure time of each MFCV and SFCV is ≤ 25 seconds on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each EFW manual, and non-automatic power operated valve in each water flow path and in the steam supply flow path to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	-------------------------NOTE------------------------ Not required to be met in MODES 3 and 4. --- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	-------------------------NOTE------------------------- Not required to be met in MODES 3 and 4. -- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Verify proper alignment of the required EFW flow paths by verifying valve alignment from the upper surge tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify the required PSW battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.2	Verify the required Keowee Hydroelectric Station power supply can be aligned to and power the PSW electrical system.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.3	Verify developed head of PSW primary and booster pumps at flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM.
SR 3.7.10.4	Verify PSW battery capacity of the required battery is adequate to supply, and maintain in OPERABLE status, required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.5	Verify the required PSW battery charger supplies ≥ 300 amps at greater than or equal to the minimum established float voltage for > 8 hours. <u>OR</u> Verify the required battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding PSW event discharge state.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.6	---------------------------NOTE--------------------------- Both HPI pump motors are individually tested although only one (1) HPI pump motor is required to support PSW system OPERABILITY. -- Verify that the required PSW switchgear and transfer switches can be aligned and power both the "A" and "B" HPI pump motors.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.7	Perform functional test of required power transfer switches used for pressurizer heaters, PSW control, electrical panels, vital I&C chargers, and valves.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.10.8	-----NOTE----- Cooling water flow to the HPI pump motors are individually tested although only flow to the HPI pump motor aligned to PSW power is required to support PSW system OPERABILITY. ----- Verify PSW booster pump and valves can provide adequate cooling water flow to HPI pump motor coolers.	In accordance with the INSERVICE TESTING PROGRAM.
SR 3.7.10.9	Verify developed head of PSW portable pump at the flow test point is greater than or equal to required developed head.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.10	Verify the required PSW valves are tested in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM.
SR 3.7.10.11	Perform CHANNEL CHECK for each required PSW instrument channel.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.12	Perform CHANNEL CALIBRATION for each required PSW instrument channel.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.10.13	Verify for the required PSW battery that the cells, cell plates and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	In accordance with the Surveillance Frequency Control Program.

3.7 Plant Systems

3.7.19 Spent Fuel Pool Cooling (SFPC) Purification System Isolation from Borated Water Storage Tank (BWST)

LCO 3.7.19

a. Two SFPC Purification System BWST automatic isolation valves shall be OPERABLE.

b. SFPC Purification System branch line manual valves shall be closed and meet INSERVICE TESTING PROGRAM leakage requirements.

APPLICABILITY: MODES 1, 2, 3 and 4 when the SFPC Purification System is not isolated from the BWST

ACTIONS

-----NOTES-----

1. SFPC Purification System flow path from the BWST may be unisolated intermittently under administrative controls.
2. Separate Condition entry allowed for each SFPC Purification System branch line manual valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One automatic isolation valve inoperable.	A.1 Isolate the flow path by use of at least one closed and de-activated automatic valve, one closed and de-activated non-automatic power operated valve, closed manual valve, or blind flange. AND A.2 Verify the flow path is isolated.	4 hours Once per 31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	Verify SFPC Purification System branch line manual valves that are not locked, sealed, or otherwise secured in position are closed.	In accordance with the Surveillance Frequency Control Program
SR 3.7.19.2	Verify SFPC Purification System branch line manual valves meet INSERVICE TESTING PROGRAM Leakage Requirements.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.19.3	Verify SFPC Purification System BWST automatic isolation valves are OPERABLE in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.19.4	Verify each SFPC Purification System BWST automatic isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.1.11	Verify for required SSF battery that the cell to cell and terminal connections are clean, tight and coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.10.1.12	Verify battery capacity of required battery is adequate to supply, and maintain in OPERABLE status, the required maximum loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program
SR 3.10.1.13	Perform CHANNEL CALIBRATION for each required SSF instrument channel.	In accordance with the Surveillance Frequency Control Program
SR 3.10.1.14	Verify OPERABILITY OF SSF valves in accordance with the INSERVICE TESTING PROGRAM.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.10.1.15	---------------------NOTE--------------------------- Not applicable to the SSF submersible pump. --- Verify the developed head of each required SSF pump at the flow test point is greater than or equal to the required developed head.	 In accordance with the INSERVICE TESTING PROGRAM
SR 3.10.1.16	Verify the developed head of the SSF submersible pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals (continued)

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, as amended by relief granted in accordance with 10 CFR 50.55a(a)(3).

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.8 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for inspection of each reactor coolant pump flywheel. At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed if the interval measured from the previous such inspection is greater than 6 2/3 years. The interval may be extended up to one year to permit inspections to coincide with a planned outage.

5.5.9 Inservice Testing Program (Deleted)

NOTE: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

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. 166
Renewed License No. NPF-63

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

    A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated November 7, 2017, 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. 166, 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 within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-63
and the Technical Specifications

Date of Issuance: August 15, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 166

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 line in the margin indicating the area of change.

Remove	Insert
NPF-63, Page 4	NPF-63, Page 4

Replace the following pages of the Appendix A Technical Specifications (TS) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
i	i
1-3	1-3
3/4 1-9	3/4 1-9
3/4 1-10	3/4 1-10
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-12	3/4 4-12
3/4 5-5	3/4 5-5
3/4 6-11	3/4 6-11
3/4 6-15	3/4 6-15
3/4 6-32	3/4 6-32
3/4 7-1	3/4 7-1
3/4 7-9	3/4 7-9
3/4 7-30	3/4 7-30
6-19g	6-19g

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. 166, 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.

[1] The 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 "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."

1.0 DEFINITIONS

$\overline{E}$ – AVERAGE DISINTEGRATION ENERGY

1.12 $\overline{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration (MeV/d) for isotopes, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

EXCLUSION AREA BOUNDARY

1.14 The EXCLUSION AREA BOUNDARY shall be that line beyond which the land is not controlled by the licensee to limit access.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

1.16 (DELETED)

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

INSERVICE TESTING PROGRAM

1.17a The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging/safety injection pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4*, 5*#, and 6*#.

ACTION:

With no charging/safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging/safety injection pump shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the reactor coolant system and reactor coolant pump seals, that a differential pressure across the pump of greater than or equal to 2446 psid is developed when tested pursuant to the INSERVICE TESTING PROGRAM.

4.1.2.3.2 All charging/safety injection pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable** by verifying that each pump's motor circuit breaker is secured in the open position prior to the temperature of one or more of the RCS cold legs decreasing below 325°F and at least once per 31 days thereafter, except when the reactor vessel head is removed.

---

* A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

** An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

For periods of no more than 1 hour, when swapping pumps, it is permitted that there be no OPERABLE charging/safety injection pump. No CORE ALTERATIONS or positive reactivity changes are permitted during this time.

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours* or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

-- NOTE --

*The 'A' Train charging/safety pump is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in the HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to the INSERVICE TESTING PROGRAM.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

---

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of the INSERVICE TESTING PROGRAM, each PORV shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Performing a CHANNEL CALIBRATION of the actuation instrumentation, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4, prior to going to 325°F.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The accumulator for the safety-related PORVs shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

d. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying automatic interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on safety injection actuation test signal and on safety injection switchover to containment sump from an RWST Lo-Lo level test signal, and

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

      a) Charging/safety injection pump,

      b) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure when tested pursuant to the INSERVICE TESTING PROGRAM:

   1. Charging/safety injection pump (Refer to Specification 4.1.2.4)

   2. RHR pump $\geq$ 100 psid at a flow rate of at least 3663 gpm.

g. By verifying that the locking mechanism is in place and locked for the following High Head ECCS throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

   2. At the frequency specified in the Surveillance Frequency Control Program.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

-- NOTE --

**The 'A' Train Containment Spray System is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position*;

b. By verifying that, on an indicated recirculation flow of at least 1832 gpm, each pump develops a differential pressure of greater than or equal to 186 psi when tested pursuant to the INSERVICE TESTING PROGRAM;

c. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

   3. Verifying that, coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal.

d. At the frequency specified in the Surveillance Frequency Control Program by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

e. At the frequency specified in the Surveillance Frequency Control Program by verifying that containment spray locations susceptible to gas accumulation are sufficiently filled with water.

---

* Not required to be met for system vent flow paths opened under administrative control.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, when tested pursuant to the INSERVICE TESTING PROGRAM.

3/4.6.5 VACUUM RELIEF SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5 The containment vacuum relief system shall be OPERABLE with an Actuation Setpoint of equal to or less negative than -2.5 inches water gauge differential pressure (containment pressure less atmospheric pressure)

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

ACTION:

With one containment vacuum relief system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more main steam line Code safety valves inoperable, operation may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by the INSERVICE TESTING PROGRAM.

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

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

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2, 3, and 4:

With one MSIV inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

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

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

-- NOTE --

*The 'A' Train Essential Services Chilled Water System loop is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.7.13 The Essential Services Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by the INSERVICE TESTING PROGRAM, and

b. At the frequency specified in the Surveillance Frequency Control Program by demonstrating that:

1. Non-essential portions of the system are automatically isolated upon receipt of a Safety Injection actuation signal, and

2. The system starts automatically on a Safety Injection actuation signal.

PROCEDURES AND PROGRAMS (Continued)

m. Inservice Testing Program (Deleted)

Note: See Section 1.17a for the definition of INSERVICE TESTING PROGRAM.

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 259
Renewed License No. DPR-23

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated November 7, 2017, 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 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 259 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-23
and the Technical Specifications

Date of Issuance: August 15, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 259

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
DPR-23, page 3	DPR-23, page 3

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
1.1-3	1.1-3
1.1-4	1.1-4
3.4-24	3.4-24
3.4-39	3.4-39
3.5-6	3.5-6
3.6-12	3.6-12
3.6-17	3.6-17
3.6-21	3.6-21
3.7-3	3.7-3
3.7-7	3.7-7
3.7-9	3.7-9
5.0-11	5.0-11

D. 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 and equipment calibration or associated with radioactive apparatus or components;

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

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 259 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

1.1 Definitions

Ē - AVERAGE DISINTEGRATION ENERGY (continued)

iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

INSERVICE TESTING PROGRAM

The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

(continued)

1.1 Definitions

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the Updated Final Safety Analysis Report (UFSAR); b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2339 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-------------------------------NOTES-------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---	
	Verify leakage from each RCS PIV is less than or equal to an equivalent of 5 gpm at an RCS pressure ≥ 2235 psig, and verify the margin between the results of the previous leak rate test and the 5 gpm limit has not been reduced by ≥ 50% for valves with leakage rates > 1.0 gpm.	In accordance with the INSERVICE TESTING PROGRAM and 24 months AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND (continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.6	Verify, by visual inspection, the ECCS containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.6.3.6	Verify each 42 inch inboard containment purge valve is blocked to restrict the valve from opening > 70°.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Operate each containment cooling train fan unit for ≥ 15 minutes.	31 days
SR 3.6.6.3	Verify cooling water flow rate to each cooling unit is ≥ 750 gpm.	31 days
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.7	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.8.3	Verify the opening time of each air operated header injection valve is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.8.4	Verify each automatic valve in the IVSW System actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.8.5	Verify the IVSW dedicated nitrogen bottles will pressurize the IVSW tank to ≥ 46.2 psig.	24 months
SR 3.6.8.6	Verify total IVSW seal header flow rate is ≤ 124 cc/minute	24 months

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	----------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift setting shall be within $\pm 1\%$.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	------------------------NOTE--------------------------------- Only required to be performed in MODES 1 and 2. --- Verify closure time of each MSIV is within limits on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more bypass valves inoperable.	C.1 Close or isolate bypass valve. AND C.2 Verify bypass valve is closed or isolated.	8 hours Once per 7 days
D Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and bypass valve is within limits on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.3.2	Verify the closure time of each MFIV is within limits on an actual or simulated actuation signal.	In accordance with the INSERVICE TESTING PROGRAM

5.5 Programs and Manuals (continued)

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program provides controls for the inspection of each reactor coolant pump flywheel in accordance with the Inservice Inspection Program.

5.5.8 Inservice Testing Program (Deleted)

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NOS. 299 AND 295 TO RENEWED FACILITY OPERATING

LICENSE NOS. NPF-35 AND NPF-52

AMENDMENT NOS. 309 AND 288 TO RENEWED FACILITY OPERATING

LICENSE NOS. NPF-9 AND NPF-17

AMENDMENT NOS. 409, 411, AND 410 TO RENEWED FACILITY OPERATING

LICENSE NOS. DPR-38, DPR-47, AND DPR-55

AMENDMENT NO. 166 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63

AMENDMENT NO. 259 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DUKE ENERGY CAROLINAS, LLC AND DUKE ENERGY PROGRESS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413, 50-414

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369, 50-370

OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By application dated November 7, 2017 (Reference 1), Duke Energy Progress, LLC, and Duke Energy Carolinas, LLC (the licensee) requested changes to the Technical Specifications (TSs) for Catawba Nuclear Station, Units 1 and 2 (Catawba); McGuire Nuclear Station, Units 1 and 2 (McGuire); Oconee Nuclear Station, Unit Nos. 1, 2, and 3 (Oconee); Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris); and H. B. Robinson Steam Electric Plant, Unit No. 2

(Robinson). The licensee requested changes to the TSs consistent with Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-545, Revision 3, "TS Inservice Testing [IST] Program Removal & [and] Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (Reference 2).

2.0 REGULATORY EVALUATION

2.1 Description of Inservice Testing Requirements and TSTF-545

An inservice test is a test to assess the operational readiness of a system, structure, or component after first electrical generation by nuclear heat. The American Society of Mechanical Engineers *Operation and Maintenance of Nuclear Power Plants* (ASME OM Code) provides requirements for inservice testing (IST) of certain components in light-water nuclear (LWR) power plants. The ASME OM Code identifies the components subject to testing (i.e., pumps, valves, pressure relief devices, and dynamic restraints), methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and record keeping. Paragraph 50.55a(f), "Inservice testing requirements," of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that IST of certain ASME Code Class 1, 2, and 3 components must meet the requirements of the ASME OM Code and applicable addenda. The TSs also prescribe IST requirements and frequencies for ASME Code Class 1, 2, and 3 components.

The regulation under 10 CFR 50.55a(f)(5)(ii) states that if a revised inservice test program for a facility conflicts with the TSs, the licensee must apply to the Commission for amendment of the TSs to conform the TSs to the revised program. Revision 3 of TSTF-545 provides guidance to licensees on how to request license amendments that would eliminate conflicting requirements between 10 CFR 50.55a and the TSs. Revision 3 of TSTF-545 proposes elimination of the IST Program from the Administrative Controls section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the IST Program. The elimination of the IST Program from the TSs could cause confusion about the correct application of these surveillance requirements. Therefore, TSTF-545, Revision 3, also proposes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs, which would be defined as the licensee program that fulfills the requirements of 10 CFR 50.55a(f). Revision 3 of TSTF-545 proposes replacement of existing uses of the term, "Inservice Testing Program," with the defined term, as denoted by capitalized letters, throughout the TSs.

By letter dated December 11, 2015 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC) found the changes to the Standard Technical Specifications (STSs) proposed in TSTF-545, Revision 3 to be suitable for incorporation into the STSs and announced that licensees could request amending their licenses to adopt TSTF-545, Revision 3. The December 15, 2015, letter also made available a model safety evaluation (SE). The NRC published a notice of availability of TSTF-545, Revision 3 in the *Federal Register* (FR) on March 28, 2016 (81 FR 17208). This SE incorporates clarifications, additional justifications, and formatting changes to the TSTF-545 model SE, consistent with other SEs for issued TSTF-545 related amendments.

2.2 Proposed Technical Specifications Changes

The licensee requested to revise the plants' TSs by deleting the IST program from the Administrative Controls TS sections for Catawba (TS 5.5.8), McGuire (TS 5.5.8), Oconee

(TS 5.5.9), Shearon Harris (TS 6.8.4.m), and Robinson (TS 5.5.8), as follows, with proposed deletions shown as stricken text and additions as bolded text.

Catawba:

Inservice Testing Program **(Deleted)**

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM. ~~This program provides controls for inservise testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:~~

a. ~~Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

b. ~~The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less for performing inservice testing activities;~~

c. ~~The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

d. ~~Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

McGuire:

Inservice Testing Program **(Deleted)**

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

~~This program provides controls for inservise testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:~~

~~a. Testing Frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d. Nothing in the ASME Boiler OM Code shall be construed to supersede the requirements of any TS.~~

Oconee:

Inservice Testing Program **(Deleted)**

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves:~~

~~a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:~~

~~ASME OM Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.~~

Shearon Harris:

Inservice Testing Program **(Deleted)**

Note: See Section 1.17a for the definition of INSERVICE TESTING PROGRAM.

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:~~

~~1) Testing frequencies specified in the ASME Code for Operations and Maintenance of Nuclear Power Plants and applicable Addenda as follows:~~

~~ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~2) The provisions of SR 4.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.~~

~~3) The provisions of SR 4.0.3 are applicable to inservice testing activities. and~~

~~4) Nothing in the ASME Code for Operation and Maintenance of Nuclear Power Plants shall be construed to supersede the requirements of any Technical Specification~~

Robinson:

Inservice Testing Program **(Deleted)**

Note: See Section 1.1 for the definition of INSERVICE TESTING PROGRAM.

~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The program shall include the following:~~

~~a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:~~

~~ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities~~	~~Required Frequencies for performing inservice testing activities~~
~~Weekly~~	~~At least once per 7 days~~
~~Monthly~~	~~At least once per 31 days~~
~~Quarterly or every 3 months~~	~~At least once per 92 days~~
~~Semiannually or every 6 months~~	~~At least once per 184 days~~
~~Every 9 months~~	~~At least once per 276 days~~
~~Yearly or annually~~	~~At least once per 366 days~~
~~Biennially or every 2 years~~	~~At least once per 731 days~~

~~b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;~~

~~c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and~~

~~d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.~~

The licensee requested to revise the Definitions section of the TSs by adding the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "Inservice Testing Program" in the plants' TSs be replaced with "INSERVICE TESTING PROGRAM," so that it refers to the new definition in lieu of the deleted program.

For Shearon Harris, the licensee proposed conforming changes to the TSs' index pages denoting the addition of the new definition.

2.3 Regulatory Requirements and Guidance

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

Technical Specifications

Paragraph 50.36(c) of 10 CFR requires TSs to include the following categories: (1) safety limits, limiting safety systems settings, and control settings; (2) limiting conditions for operation;

(3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Paragraph 50.36(c)(3) states that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Paragraph 50.36(c)(5) states that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The NRC staff's guidance for review of the TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 16, "Technical Specifications," Revision 3, March 2010 (Reference 4). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared improved STSs for each of the LWR nuclear steam supply systems and associated balance-of-plant equipment systems. The licensee's proposed amendments are based on TSTF-545, Revision 3, which is an NRC-approved change to the improved STSs. The NRC staff's review included consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. The NRC staff gives special attention to TS provisions that depart from the improved STSs, as modified by NRC-approved TSTF travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include IST of pumps and valves at nuclear power reactors in accordance with the ASME OM Code as specified in 10 CFR 50.55a(f). Section 50.55a(f) states in-part that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel (BPV) Code and ASME OM Code as specified in the paragraph, and that each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions of 10 CFR 50.55a(f)(1) through (f)(6).

The ASME OM Code is a consensus standard, which is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code IST program requirements were suitable for incorporation into the NRC's rules.

Paragraph 50.55(a)(f)(5)(ii) of 10 CFR states in part that if a revised inservice test program for a facility conflicts with the TSs for the facility, the licensee must apply for an amendment of the TSs to conform the TSs to the revised program.

Revision 2 of NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants," Final Report, October 2013 (Reference 5), provides guidance for the IST of pumps and valves.

NUREG-0800, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, March 2007 (Reference 6), provides guidance and acceptance criteria for the NRC staff's review of the IST program for pumps and valves.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the regulations, guidance, and licensing information discussed in Section 2.3 of this SE. 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. Among the considerations are whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5) (i.e., provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner). The NRC staff also considered whether the TSs, as amended, would assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met per 10 CFR 50.36(c)(3). In making its determination as to whether to amend the license, the NRC staff considered those regulatory requirements that are automatically conditions of the license through 10 CFR 50.54. Where the regulations already condition the license, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Deletion of the Inservice Testing Program Technical Specifications

For Catawba, McGuire, Oconee, Shearon Harris, and Robinson, the IST program TSs are TS 5.5.8, TS 5.5.8, TS 5.5.9, TS 6.8.4.m, and TS 5.5.8, respectively, which are in the Administrative Controls section of the TSs. The IST program TSs have requirements for IST of ASME Code Class 1, 2, and 3 components (i.e., pumps and valves). Through 10 CFR 50.54, the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. These requirements include 10 CFR 50.55a(f), which specifies the requirements for the IST of pumps and valves. Therefore, requiring the licensee to have an IST program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an IST program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained further in this SE, it is not necessary to have additional administrative controls in the TSs for these plants relating to the IST program to assure operation of the facility in a safe manner.

Deletion of Technical Specification Inservice Testing Program Frequency Descriptions

The ASME OM Code requires testing to normally be performed within certain time periods. The plants' current TS IST program proposed to be deleted set IST frequencies more precisely than those specified in the ASME OM Code and applicable addenda (e.g., "at least once per 31 days" contrasted with "monthly"). However, the NRC staff determined that the more precise IST frequencies are not necessary to assure operation of the facility in a safe manner. Therefore, the NRC staff finds these proposed changes acceptable.

Deletion of Surveillance Requirement 3.0.2/4.0.2 Provisions from Inservice Testing Program Technical Specifications

The plants' current TS IST program proposed to be deleted allows the licensee to use SR 3.0.2 (SR 4.0.2 for Shearon Harris) to extend, by up to 25 percent, the interval between IST activities, as required by the plants' IST TSs. Similar to these TSs, Code Case OMN-20, incorporated by reference in 10 CFR 50.55a, also permits the licensee to extend the IST intervals specified in

the ASME OM Code by up to 25 percent. The NRC staff determined that the TS allowance to extend IST intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of these TSs is acceptable. Therefore, the NRC staff finds these proposed changes acceptable.

Deletion of Surveillance Requirement 3.0.3/4.0.3 Provisions from Inservice Testing Program Technical Specifications

The plants' current TS IST program proposed to be deleted allows the licensee to use SR 3.0.3 (SR 4.0.3 for Shearon Harris) when it discovers that an SR associated with an inservice test was not performed within its specified frequency. Surveillance Requirement 3.0.3 (or SR 4.0.3) allows the licensee to delay declaring a limiting condition for operation not met in order to perform the missed surveillance. The use of SR 3.0.3 (or SR 4.0.3) for inservice tests is limited to those inservice tests required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. Deletion of this TS does not change any of these requirements, and SR 3.0.3 (or SR 4.0.3) will continue to apply to those inservice tests required by SRs. Therefore, the NRC staff finds the deletion of these TSs is acceptable.

Deletion of Catawba TS 5.5.8d, McGuire TS 5.5.8d, Oconee TS 5.5.9d, Shearon Harris TS 6.8.4.m4, and Robinson TS 5.5d.8.

Catawba TS 5.5.8d, McGuire TS 5.5.8d, Oconee TS 5.5.9d, Shearon Harris TS 6.8.4.m4, and Robinson TS 5.5d.8 state that nothing in the ASME OM (or BPV) Code shall be construed to supersede the requirements of any TS. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised IST program for a facility conflicts with the TSs for the facility. These regulations require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the IST program because the regulations specify how conflicts must be resolved. Therefore, the NRC staff finds the deletion of these TSs acceptable.

Conclusion Regarding Deletion of Inservice Testing Program Technical Specifications

The NRC staff determined that the requirements currently in the IST program TSs are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the NRC staff concludes that deletion of the IST program TSs from the licensee's TSs is acceptable because the IST program TSs are not required by 10 CFR 50.36(c)(5).

3.2 Definition of INSERVICE TESTING PROGRAM and Revision to Surveillance Requirements

The licensee proposed to revise the TS Definitions section to include the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition is consistent with the definition in TSTF-545, Revision 3. The NRC staff finds the definition acceptable because it correctly refers to the IST requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "Inservice Testing Program" in the TSs be revised to "INSERVICE TESTING PROGRAM" to reference the new TS defined term in lieu

of the deleted IST program TSs. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in TSs with the new definition. The NRC staff verified that for each reference to the "Inservice Testing Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the SR testing is performed. However, the IST frequencies could change because the plants' TSs will no longer include the more precise test frequencies. As discussed in Section 3.1 of this SE, the NRC staff determined that the TSs do not need to include the more precise testing frequencies currently in the TSs. Based on its review, the NRC staff determined that revising the SRs to refer to the new definition is acceptable because these SRs will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The NRC staff also determined that with the proposed changes, 10 CFR 50.36(c)(3) will continue to be met because the SRs will continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

3.3 Conforming Changes and Variations from TSTF-545

The NRC staff evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this SE.

a. The TSs for Shearon Harris have not been converted to the improved STSs on which TSTF-545, Revision 3 is based. As a result, the numbering, format, and content of these TSs vary from TSTF-545, Revision 3. In addition, all the plants' TSs use different numbering than the improved STSs, on which TSTF-545, Revision 3 is based. The NRC staff finds that the licensee's proposed differences in numbering, format, and content are editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the NRC staff finds that the licensee's proposed TS changes are acceptable.

b. The index for Shearon Harris TSs is included as part of the TSs. Therefore, the licensee included conforming changes to the index resulting from the addition of the new definition. The NRC staff finds that the proposed differences are editorial in nature and that the licensee's proposed TS changes are consistent with the intent of TSTF-545, Revision 3. Therefore, the NRC staff finds that the licensee's proposed TS changes are acceptable.

c. The IST program in the TSs for Robinson refers to testing frequencies specified in Section XI of the ASME BPV Code, which varies from TSTF-545, Revision 3. The Code of record for Robinson's IST is the ASME OM Code, and 10 CFR 50.55a(f) requires the IST program to meet the ASME OM Code. Therefore, deletion of this TSs does not create new requirements for this plant. As discussed in Section 3.1 of this SE, the NRC staff finds the proposed deletion of the IST program TSs, which include these references to Section XI of the ASME BPV Code, acceptable.

d. The licensee proposed to replace the content of IST program TSs for all plants with the word, "Deleted," and retain the existing numbering sequence. The licensee also added a note referring to the new IST definition. The NRC staff finds that these proposed changes are editorial in nature and consistent with the

intent of TSTF-545, Revision 3. Therefore, the NRC staff finds that the licensee's proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified officials from the States of North Carolina and South Carolina on June 21, 2018, of the proposed issuance of the amendments. Each State's official had no comments.

5.0 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 SRs. 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 the finding published in the *Federal Register* on January 16, 2018 (83 FR 2227). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Henderson, K., Duke Energy Progress, LLC and Duke Energy Carolinas, LLC, letter to U.S. Nuclear Regulatory Commission, "Application to Revise Technical Specifications to Adopt TSTF-545, Revision 3, 'TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing,'" dated November 7, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17312A362).

2. Technical Specifications Task Force, Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing," dated October 21, 2015 (ADAMS Accession No. ML15294A555).

3. Hsueh, K., U.S. Nuclear Regulatory Commission, letter to Technical Specifications Task Force, "Final Model Safety Evaluation of Technical Specifications Traveler TSTF-545, Revision 3, 'TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing' (TAC No. MF3349)," dated December 11, 2015 (ADAMS Package Accession No. ML15317A071).

4. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 16, "Technical Specifications," Revision 3, March 2010 (ADAMS Accession No. ML100351425).

5. U.S. Nuclear Regulatory Commission, NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints (Snubbers) at Nuclear Power Plants," Revision 2, October 2013 (ADAMS Accession No. ML13295A020).

6. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," Revision 3, March 2007 (ADAMS Accession No. ML070720041).

Principal Contributors: Caroline Tilton

Date: August 15, 2018

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2; MCGUIRE NUCLEAR STATION, UNITS 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENTS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (EPID L-2017-LLA-0377) DATED AUGUST 15, 2018

DISTRIBUTION:
PUBLIC
PM File Copy
RidsACRS_MailCTR Resource
RidsNrrDorlLpl2-1 Resource
RidsNrrDorlLpl2-2 Resource
RidsNrrDssStsb Resource
RidsNrrLABClayton Resource
RidsNrrLAJBurkhardt Resource
RidsNrrLAKGoldstein Resource
RidsNrrPMCatawba Resource
RidsNrrPMShearonHarris Resource
RidsNrrPMMcGuire Resource
RidsNrrPMOconee Resource
RidsNrrPMRobinson Resource
RidsRgn2MailCenter Resource
CTilton, NRR

ADAMS Accession No.: ML18172A172 * by memo ML18170A301 ** by email

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DSS/STSB/BC*
NAME	DGalvin	BClayton (JBurkhardt for)	VCusumano
DATE	7/30/18	7/25/18	6/20/18
OFFICE	OGC **	NRR/DORL/LPL2-2/BC(A)	NRR/DORL/LPL2-2/PM
NAME	BHarris (NLO)	BVenkataraman	DGalvin
DATE	8/9/18	8/15/18	8/15/18

OFFICIAL RECORD COPY