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

March 31, 2020

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

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 176 REGARDING THE EXTENSION OF THE ESSENTIAL SERVICES CHILLED WATER SYSTEM ALLOWED OUTAGE TIME AND REMOVAL OF AN EXPIRED NOTE FROM TECHNICAL SPECIFICATIONS (EPID L-2019-LLA-0025)

Dear Ms. Maza:

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 176 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment revises Technical Specifications (TSs) requirements in response to your application dated February 18, 2019, as supplemented by letters dated September 3, 2019, and November 21, 2019.

The amendment allows one train of the Essential Services Chilled Water System (ESCWS) to be inoperable for up to 7 days from the currently allowed 72 hours for extended maintenance activities on the ESCWS and air handlers supported by the ESCWS for equipment reliability. The amendment would also remove an expired Note in numerous TS sections that was previously added by implementation of License Amendment No. 153.

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

Sincerely,

/RA/

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

Docket No. 50-400

Enclosures:
1. Amendment No. 176 to NPF-63
2. Safety Evaluation

cc: Listserv

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. 176
Renewed License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated February 18, 2019, as supplemented by letters dated September 3, 2019, and November 21, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 176, 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: March 31, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 176

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

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

Remove	Insert
Page 4	Page 4

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

Remove	Insert
3/4 1-10	3/4 1-10
3/4 5-3	3/4 5-3
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
3/4 7-4	3/4 7-4
3/4 7-11	3/4 7-11
3/4 7-12	3/4 7-12
3/4 7-14	3/4 7-14
3/4 7-17	3/4 7-17
3/4 7-30	3/4 7-30
3/4 8-2	3/4 8-2

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. 176, 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 of “Carolina Power & Light Company” (CP&L) was changed to “Duke Energy Progress, Inc.” On August 1, 2015, the name “Duke Energy Progress, Inc.” was changed to “Duke Energy Progress, LLC.”

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----------

*One charging/safety injection pump train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

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.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

----------NOTE----------

*One ECCS subsystem train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

1. Verifying that the following valves are in the indicated positions with the control power disconnect switch in the "OFF" position, and the valve control switch in the "PULL TO LOCK" position:

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----------

**One Containment Spray System train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

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.

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A Spray Additive Tank containing a volume of between 3268 and 3768 gallons of between 27 and 29 weight % of NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive 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. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

c. At the frequency specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a containment spray or containment isolation phase A test signal as applicable; and

d. At the frequency specified in the Surveillance Frequency Control Program by verifying each eductor flow rate is between 17.2 and 22.2 gpm, using the RWST as the test source containing at least 436,000 gallons of water.

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3, and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH- 4.

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

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers 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. Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one train of the above required containment fan coolers inoperable and 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 and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

----------NOTE----------

*One train of containment fan coolers and one Containment Spray System train are allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

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

1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

2. Verifying a cooling water flow rate, after correction to design basis service water conditions, of greater than or equal to 1300 gpm to each cooler.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that each fan train starts automatically on a safety injection test signal.

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. (NOTE: LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. Following restoration of one AFW train, all applicable LCOs apply based on the time the LCOs initially occurred.)

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

1. Demonstrating that each motor-driven pump satisfies performance requirements by either:

a) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1514 psid at a recirculation flow of greater than or equal to 50 gpm (25 KPPH), or

b) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1259 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH).

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

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

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths 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

4.7.3 At least two component cooling water flow paths 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) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

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

1. Each automatic valve servicing safety-related equipment or isolating non-safety-related components actuates to its correct position on a Safety Injection test signal, and

2. Each Component Cooling Water System pump required to be OPERABLE starts automatically on a Safety Injection test signal.

3. Each automatic valve serving the gross failed fuel detector and sample system heat exchangers actuates to its correct position on a Low Surge Tank Level test signal.

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* The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent emergency service water loops shall be OPERABLE.

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

ACTION:

With only one emergency service water 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 'B' Train emergency service water loop is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two emergency service water loops 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) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

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

   1. Each automatic valve servicing safety-related equipment or isolating non-safety portions of the system actuates to its correct position on a Safety Injection test signal, and

   2. Each emergency service water pump and each emergency service water booster pump starts automatically on a Safety Injection test signal.

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Filtration System (CREFS) trains shall be OPERABLE.*

APPLICABILITY:

a. MODES 1, 2, 3, and 4

b. MODES 5 and 6

c. During movement of irradiated fuel assemblies and movement of loads over spent fuel pools

ACTION:

a. MODES 1, 2, 3 and 4:

----------NOTE----------
In addition to the Actions below, perform Action c. if applicable.
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1. With one CREFS train inoperable for reasons other than an inoperable Control Room Envelope (CRE) boundary, restore the inoperable CREFS train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more CREFS trains inoperable due to inoperable CRE boundary:

   a. Initiate action to implement mitigating actions immediately or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours;

   b. Within 24 hours, verify mitigating actions ensure CRE occupant radiological exposures will not exceed limits and that CRE occupants are protected from hazardous chemicals and smoke or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours;

   c. Restore CRE boundary to OPERABLE within 90 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* The control room envelope (CRE) boundary may be opened intermittently under administrative controls.

3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent RAB Emergency Exhaust Systems shall be OPERABLE.*

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

ACTION:

a. With one RAB Emergency Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two RAB Emergency Exhaust Systems inoperable due to an inoperable RAB Emergency Exhaust System boundary, restore the RAB Emergency Exhaust System boundary to OPERABLE status within 24 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each RAB Emergency Exhaust System shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 continuous minutes;

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the unit flow rate is 6800 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980;

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, has a methyl iodine penetration of ≤ 2.5% when tested at a temperature of 30°C and at a relative humidity of 95% in accordance with ASTM D3803-1989.

* The RAB Emergency Exhaust Systems boundary may be opened intermittently under administrative controls.

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 7 days* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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.

*Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

3. Restore the diesel generator 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; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

c. With one offsite circuit and one diesel generator of 3.8.1.1 inoperable:

NOTE: Enter applicable Condition(s) and Required Action(s) of LCO 3/4.8.3, ONSITE POWER DISTRIBUTION - OPERATING, when this condition is entered with no A.C. power to one train.

1. Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Following restoration of one A.C. source (offsite circuit or diesel generator), restore the remaining inoperable A.C. source to OPERABLE status pursuant to requirements of either ACTION a or b, based on the time of initial loss of the remaining A.C. source.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated February 18, 2019 (Reference 1), as supplemented by letters dated September 3, 2019 (Reference 2) and November 21, 2019 (Reference 3), Duke Energy Progress, LLC (the licensee) submitted a license amendment request (LAR) for changes to the Shearon Harris Nuclear Power Plant, Unit 1 (Harris), Technical Specifications (TSs). The requested changes would permanently modify TSs to permit one train of the essential services chilled water system (ESCWS) to be inoperable for up to 7 days from the currently allowed 72 hours for extended maintenance activities on the ESCWS and air handlers supported by the ESCWS for equipment reliability. The amendment would also remove an expired Note in numerous TS sections that was previously added by implementation of License Amendment No. 153.

The supplemental letters dated September 3, 2019 and November 21, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on July 2, 2019 (84 FR 31632).

2.0 REGULATORY EVALUATION

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

2.1 System Description

Harris' Updated Final Safety Analysis Report (UFSAR), Section 9.2.8, "Essential Services Chilled Water System," states that the ESCWS consists of two 100 percent capacity subsystems (one is normally operating, and one is normally in standby). Each subsystem consists of a package water chiller, a chilled water pump, an expansion tank, a makeup tank, a chemical addition tank, service water recirculation pump, and an independent piping system.

The ESCWS provides chilled water to the cooling coils of air handling units for the following systems:

- Control Room Air Conditioning System
- Reactor Auxiliary Building (RAB) Engineered Safety Feature (ESF) Equipment Cooling System
- RAB Switchgear Rooms Ventilation System
- RAB Electrical Equipment Protection Rooms Ventilation System
- RAB Non-Nuclear Safety (NNS)-Ventilation Systems
- Fuel Handling Building Spent Fuel Pool Pump Room Ventilation System

The ESCWS is automatically started upon receipt of a safety injection actuation signal (SIAS). Non-essential portions of the ESCWS are automatically isolated from the essential portions upon receipt of a SIAS. In the event of a failure in a single train of the ESCWS during an accident, a redundant 100 percent capacity system would still be available. Upon receipt of a SIAS, the demineralized water supply to the chillers will be isolated using redundant solenoid operated valves arranged in series. The supply will then be provided from the service water system (SWS). The source of water supply to the condenser section of the ESCWS is from the SWS during normal and emergency plant conditions.

In the event of loss of offsite power, all active components such as valve operators, water chiller motors, chilled water pumps, controls and instrumentation will be supplied with power from the emergency diesel generators (EDGs). Each subsystem is powered from a different emergency bus. Upon loss of offsite power, the ESCWS chillers and chilled water pumps are automatically sequenced to reduce starting power requirements from the standby EDG. Each chiller is furnished with a compressor starter, operational and safety controls, interlocks and other controls for local and remote operation.

2.2 Description of the Proposed Changes

In Section 2 of Attachment 1 to the LAR, the licensee provides a detailed description of the proposed changes. The following six parts of the TS are involved in the proposed changes:

(1) TS 3.1.2.4, “Charging Pumps – Operating”
(2) TS 3.5.2, “ECCS [Emergency Core Cooling System] Subsystems – T_{avg} Greater Than or Equal to 350°F”
(3) TS 3.6.2.1, “Containment Spray System”
(4) TS 3.6.2.3, “Containment Cooling System”
(5) TS 3.7.4, “Emergency Service Water System”
(6) TS 3.7.13, “Essential Services Chilled Water System”

The proposed license amendment revises TS 3.1.2.4, TS 3.5.2, TS 3.6.2.1, TS 3.6.2.3, TS 3.7.4 that will apply for ‘B’ Train completion time (CT) extensions only, and TS 3.7.13 which requires at least two independent ESCWS loops to be operable in modes 1-4. These systems and components all have redundant trains. The current TS allows one train to be inoperable up to 72 hours or in hot standby condition after that period. In the past 5-years, there were a few instances where maintenance activities lasted 41-57 hours of the allowed 72 hours. The licensee requests the CT extension for flexibility of completing maintenance activities and installation of planned ESCWS modification-related maintenance activities for equipment reliability.

While the LAR is for maintenance on the ESCWS and air handlers supported by the ESCWS, the impact of this support system on other TS systems associated with the inoperable train is accounted for in the proposed TS changes. Under the proposed changes, an inoperable train of the ESCWS must be restored to operable status within 7 days. The overall reliability is reduced because a single failure to the operable train could result in a loss of function. The licensee provides the compensatory measures in Section 3.6 of Attachment 1 to the LAR to manage risk during the proposed CT and the probabilistic risk assessment (PRA) analysis and calculation in Attachment 5 to the LAR to evaluate the impact on plant risk.

The licensee proposed allowing 7 days for the ESCWS and equipment supported by ESCWS to be inoperable during maintenance on the ESCWS and air handlers supported by the ESCWS. Proposed footnotes in the affected supported system TS would state that the 7-day allowance is only applicable if compensatory measures described in the LAR are implemented prior to exceeding 72 hours of inoperability. In the supplement dated November 21, 2019, the licensee proposed the following footnote for supported systems:

> *One [system name] train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

Where the bracketed [system name] would contain the name of the system supported by the ESCWS for the respective TS. The footnotes were proposed for TS 3.1.2.4, 3.5.2, 3.6.2.1, 3.6.2.3, and 3.7.4.

Likewise, in the supplement dated November 21, 2019, the licensee proposed changing the allowed operating time for the situation where only one ESCWS is OPERABLE from 72 hours to "7 days*" in TS 3.7.13, with the following corresponding footnote:

> *Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

The licensee also proposed deleting expired footnotes from TS 3.1.2.4, TS 3.5.2, TS 3.6.2.1, TS 3.6.2.2, TS 3.6.2.3, TS 3.7.1.2, TS 3.7.3, TS 3.7.4, TS 3.7.6, TS 3.7.7, TS 3.7.13 and TS 3.8.1.1.

2.3 Regulatory Requirements

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for licenses to operate nuclear power plants to include TSs as part of the license application. These TSs become part of any license issued.

The regulation at Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.90 to 10 CFR Part 50 states, in part, that whenever a holder of a license desires to amend the license, application for an amendment must be filed fully describing the changes desired including technical specifications in the license. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in

10 CFR 50.57(a)(3), provide that there must be “reasonable assurance” that the activities at issue will not endanger the health and safety of the public, and will comply with the NRC’s regulations.

Section 50.36(a)(1) to 10 CFR Part 50 states, in part, that an applicant for an operating license include in the application, proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must also include in the application a “summary statement of the bases or reasons for such specifications, other than those covering administrative controls.” However, per 10 CFR 50.36(a)(1), these TS bases “shall not become part of the technical specifications.”

Section 50.36(b) to 10 CFR Part 50 states, in part, that the TSs to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto. As required by 10 CFR 50.36(c)(2), the technical specifications include limiting conditions for operation, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. The remedial actions must provide the requisite “reasonable assurance” of safety and compliance.

Section 50.36(c)(2) to 10 CFR Part 50 states, in part, that there are four criteria used to determine whether or not LCOs must be established in the TSs for items related to plant operation. If the item meets one or more of the four criteria listed below, an LCO must be established in the TSs to ensure the lowest functional capability or performance level of equipment required for safe operation of the facility will be met. The four criteria are:

> Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4. A structure, system, or component which operating experience or PRA has shown to be significant to public health and safety.

Section 50.36(c)(5) to 10 CFR Part 50 states, in part, 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 regulations in Appendix A to 10 CFR Part 50, “General Design Criteria for Nuclear Power Plants” (hereinafter referred to as GDC), establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety.

The NRC staff identified the following GDCs applicable to this LAR:

- GDC 2, “Design Bases for Protection Against Natural Phenomena,” which requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

- GDC 4, “Environmental and Dynamic Effects Design Bases,” which requires, in part, that SSCs important to safety shall be designed and appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 44, “Cooling Water,” which requires, in part, that a system to transfer heat from SSCs important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating and accident conditions.

- GDC 45, “Inspection of Cooling Water System,” which requires, in part, that the cooling water system shall be designed to permit appropriate periodic inspection of important components to assure the integrity and capability of the system.

- GDC 46, “Testing of Cooling Water System,” which requires, that the cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

2.4 Applicable Guidance

The guidance that the NRC staff considered in its review of this LAR includes the following.

The Harris TS are based on earlier guidance for TS format and content, NUREG-0452, “Standard Technical Specifications for Westinghouse Pressurized Water Reactors,” Revision 4 (Reference 4). Licensees are not required to adopt the most current guidance. NUREG-1431, “Standard Technical Specifications, Westinghouse Plants, Specifications,” Revision 4.0 (Reference 5), provides the current version of improved STS for Westinghouse Plants. The abstract for NUREG-1431 states, in part, that licensees are encouraged to upgrade their TSs consistent with the criteria and conforming, to the practical extent, to Revision 4.0 to the improved STS.

Regulatory Guide (RG) 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” (Reference 6) describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk-acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 7), describes an acceptable risk-informed approach for assessing TS changes, specifically changes to CTs. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. Section C.2.4 of RG 1.177 provides the following three-tiered TS acceptance guidelines for evaluating the risk associated with the permanent CT changes:

1. The licensee has demonstrated that TS CT change has only a small quantitative impact on plant risk. An incremental conditional core damage probability (ICCDP) of less than 1.0×10^{-6} and an incremental conditional large early release probability (ICLERP) of less than 1.0×10^{-7} are considered small for a single TS condition entry. (Tier 1)

2. The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change. (Tier 2)

3. The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program. (Tier 3)

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 8), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors.

Relevant sections of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (hereinafter referred to as the SRP) used in the review of this LAR include the following:

- SRP Section 16.1, Revision 1, "Risk-Informed Decision Making: Technical Specifications," (Reference 9) which includes changes of allowed outage time (AOT), or its equivalent terminology CT as noted in SRP 16.1, as part of risk-informed decision-making.

- SRP Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," (Reference 10) provides guidance for evaluating PRA acceptability.

- SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," (Reference 11) provides general guidance for evaluating the technical basis for proposed risk-informed changes.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine whether the proposed TS changes are consistent with the regulations, licensing and design basis information, and regulatory guidance discussed in Section 2 of this safety evaluation.

The LAR states the change in risk associated with the proposed TS CT extension of the ESCWS was evaluated in accordance with the guidance of RG 1.174, Revision 3 and RG 1.177, Revision 1. RG 1.177 describes a risk-informed approach acceptable to the NRC, for assessing proposed changes to TS CTs, which is based on meeting the five key principles outlined in RG 1.174 and summarized in Section 2 of this SE. The NRC staff reviewed the proposed extension of the ESCWS TS CT against the five key principles of RG 1.174:

- Principle 1: The proposed licensing basis change meets the current regulations, unless it is explicitly related to a requested exemption.

- Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.

- Principle 3: The proposed licensing basis change maintains sufficient safety margins.

- Principle 4: When proposed licensing basis changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

- Principle 5: The impact of the proposed licensing basis change should be monitored using performance measurement strategies.

3.1 Principle 1: Compliance with Current Regulations

As a key principle of risk-informed integrated decision-making, Regulatory Position 1 in RG 1.174 states, the licensee should affirm that the proposed licensing basis change meets the current regulations, unless the proposed change is explicitly related to a proposed exemption (i.e., a specific exemption under 10 CFR 50.12).

In Section 3.6 of Attachment 1 to the LAR, the licensee provided a list of compensatory measures that the licensee will implement prior to exceeding 72 hours from the time of TS 3.7.13 LCO entry. The justification for the proposed temporary CT relies on the following compensatory measures:

1. The following equipment and the corresponding power supplies will be posted protected:

    - Air handling units for the operable charging safety injection pump (CSIP) areas:
        - AH-9A (CSIP 1A-SA Area),
        - AH-9B (CSIP 1B-SB Area),
        - or AH-10 (CSIP 1C-SAB Area)
    - Air handling units for the Switchgear Rooms with operable equipment:
        - AH-12 1A-SA and AH-12 1B-SA supply Switchgear Room A;
        - AH-13 1A-SB and AH-13
        - 1B-SB supply Switchgear Room B
    - Operable ESCWS chiller and operable chilled water pump

2. The Fire Protection tracking log will be reviewed for fire hazards and fire impairments. Transient combustibles and hot work in these fire risk-sensitive areas will be limited:

    - Fire compartment FC25 – RAB heating, ventilation, and air conditioning (HVAC) Room (Motor Control Center (MCC) 1A21-SA, MCC 1A31-SA)

    - Fire compartments FC34 and FC35 – Switchgear Room A and Switchgear Room B

    - Fire compartment FC41 – Turbine Building (Zone 1-G-261 – 6.9 kV Switchgear)

    - Fire compartment FC54 – Transformer Yard

3. Restrictions on work activities will be in place that involve components that if lost or failed could result in a direct plant trip or transient.

4. Operator actions for the CSIP area cooling, Switchgear Room cooling, and Auxiliary Relay Cabinet Room cooling, if needed following a loss of HVAC, will be briefed with Operations. The fan used for the CSIP area cooling will be pre-staged and verified to be functional.

5. Discretionary maintenance or discretionary testing on equipment that support the following systems will be avoided for the remaining duration of the TS 3.7.13 LCO entry:

    - ESCWS (operable train)
    - Motor-Driven and Turbine-Driven Auxiliary Feedwater Pumps
    - Emergency Service Water System and normal Service Water System
    - EDGs
    - ASI [Alternate Seal Injection] System and DSDG [Dedicated Shutdown Diesel Generator]

The NRC staff reviewed the compensatory measures and noted that implementation of compensatory actions of blocking selected doors open may degrade required protective barriers to those rooms. However, the licensee indicates in its letter dated September 3, 2019, that the compensatory action to open the doors between the Main Control Room and the Auxiliary Relay Cabinet Room has no effect on PRA component operability, and that thus, the compensatory action is provided only as an additional level of defense-in-depth. The current Harris TSs require that when one or more required barrier is unable to perform their required function(s) LCO 3.0.6 is entered for that barrier. While in the CT, failure of the remaining ESCWS train would place the plant outside of the limiting condition for operation for TS 3.7.13. The loss of both trains of ESCWS will place the unit in LCO 3.0.3 which requires initiating a shutdown sequence directed toward COLD SHUTDOWN within 36 hours. TS 3.0.3, requiring the unit be placed in at least HOT STANDBY within 6 hours, at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours. These requirements are not changed by the LAR.

Since the proposed changes would allow 7 days of continued plant operation when only one ESCW train is OPERABLE, compared to the existing 72-hour allowance, the NRC staff determined that the proposed extended CT is a relaxation to existing TS requirements. The NRC staff determined that the relaxation is acceptable because the respective actions in the TS of the affected systems, combined with the compensatory measures, are acceptable remedial measures when the respective LCO is not met during maintenance on the ESCWS and air handlers supported by the ESCWS. The NRC staff further determined the changes maintain adequate assurance of safety when evaluated against current regulatory standards.

The NRC staff has reviewed the licensee's evaluation and justification of the proposed change. The NRC staff reviewed the proposed deletion of the expired footnotes and determined the deletions are acceptable because the footnotes were temporary and can no longer be applied to the associated TS. The NRC staff reviewed the new proposed footnotes which represent a risk-informed LAR. One acceptable approach for making risk-informed decisions about proposed TS changes is to show that the proposed changes meet the five key principles stated in RG 1.177.

In Section 3.7 of Attachment 1 to the LAR, the licensee states that this LAR does not propose to deviate from existing regulatory requirements. Compliance with existing regulations is maintained by the proposed changes to the plant's TS requirements, as discussed in Section 4.1 of Attachment 1 to the LAR.

The ESCWS operating during a CT is designed to transfer heat from the SSCs important to safety it serves independent of the system out-of-service. Extending the CT from 72 hours to 7 days does not alter the design heat removal capacity from the operating train of ESCWS. The NRC staff has reviewed the licensee's existing design and finds that the existing design capability for periodic inspection of important components is not altered, nor is the existing design for cooling water system testing altered by this amendment.

Attachment 2 to the licensee's submittal dated February 18, 2019, provided revised TS Bases pages to be implemented with the associated TS changes. The NRC staff determined that TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," dated July 2, 1993 (58 FR 39132).

3.2 Principle 2: Evaluation of Defense-in-Depth

Defense-in-depth is an approach to designing and operating nuclear facilities involving multiple independent and redundant layers of defense to compensate for human and system failures. Regulatory Position C.2.1.1 in RG 1.174 states that defense-in-depth consists of seven elements and consistency with the defense-in-depth philosophy is maintained if the following occurs:

1. Preserve a reasonable balance among the layers of defense.

2. Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

4. Preserve adequate defense against potential common cause failures (CCFs).

5. Maintain multiple fission product barriers.

6. Preserve sufficient defense against human errors.

7. Continue to meet the intent of the plant's design criteria.

In Section 3.5 of Attachment 1 to the LAR, the licensee provides a discussion regarding each of these seven items. The licensee discusses how its risk informed assessment is consistent with the philosophy of defense-in-depth. The following sections provide the NRC staff's evaluation of each of the seven considerations.

3.2.1 Preserve a reasonable balance among the layers of defense

In Section 3.5 (Item 1) of Attachment 1 to the LAR, the licensee discusses the reasonable balance among the layers of defense. The licensee performed a room heatup analysis to determine the expected area temperatures in the RAB when the HVAC systems are not functioning. The analysis reviewed 19 areas including the Residual Heat Removal system/Containment Spray rooms, CSIP rooms, Component Cooling Water system area, Switchgear Rooms, Battery Rooms, Auxiliary Relay Cabinet Room, and the Main Control Room. The analysis screened the rooms for unacceptable temperatures resulting from a loss of HVAC event based on the determination of the maximum temperature in each area within 24 hours following a loss of HVAC event. The resulting temperatures were evaluated for acceptability based upon industry standards for equipment temperature limits, equipment manufacturer qualification packages, and equipment qualification temperature values. The initial screening conservatively assumes a loss of HVAC in all rooms modeled. The highest heat loads from the analysis were utilized for screening purposes.

Special consideration was given to failure of the Control Room Complex HVAC, since increased temperatures in relay and instrumentation cabinets could result in failure of these components and affect control of the plant. The Control Room Complex is continuously manned, so HVAC malfunctions would be immediately noticed, either by redundant alarms or by physically noticing the change in temperature. In the event of an HVAC malfunction, compensatory actions such as placing temporary fans, opening cabinet doors, taking manual control of components or using local indications would occur. Based on this consideration, a loss of HVAC to the Control Room Complex is not expected to significantly increase the overall risk of severe accidents.

The NRC staff notes that the proposed changes do not significantly affect the availability and reliability of SSCs that mitigate accident conditions and that the proposed changes do not significantly reduce the effectiveness of the licensee's emergency preparedness program. The NRC staff's review notes that the licensee used the highest heat loads from the heat up analysis to determine equipment operation, which represents conservatism.

The NRC staff has reviewed the licensee's analysis provided in Section 3.5 of Attachment 1 of its submittal dated February 18, 2019, as supplemented, and finds that the proposed changes continue to preserve a reasonable balance between prevention of core damage, prevention of

containment failure, and consequence mitigation. The NRC staff determined that for those rooms which could exceed acceptable temperature limits following a loss of HVAC, the pre-staging of portable ventilation equipment as required within the first 72 hours along with the compensatory actions taken for those rooms following a loss of HVAC will maintain those rooms within acceptable temperature limits.

3.2.2 Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures

In Section 3.5 (Item 2) of Attachment 1 to the LAR, the licensee discusses the adequate capability of design features for the ESCWS. The proposed changes will not allow plant operation in a configuration outside the design basis. While the compensatory measures listed in Section 3.6 of Attachment 1 to the LAR are not credited in the PRA analysis, nor are they the basis for requesting the proposed changes, these compensatory measures are intended to manage risk, and therefore, further reduce the risk of any potential risk-significant configurations during the proposed CT.

3.2.3 Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty

In Section 3.5 (Item 3) of Attachment 1 to the LAR, the licensee discusses the redundancy of the ESCWS and its related systems. The licensee also stated that the PRA analysis for this LAR indicates that the proposed CT extension provides acceptable system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty. The NRC staff has reviewed the licensee's submittal dated February 18, 2019 and the supplemental information provided in letters dated September 3, 2019, and November 21, 2019. The NRC staff finds that system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties because the proposed change does not introduce new system dependencies.

3.2.4 Preserve adequate defense against potential CCFs

In Section 3.5 (Item 4) of Attachment 1 to the LAR, the licensee states that there is no change in failure mechanisms associated with the ESCWS as a result of the CT change from 72 hours to 7 days. In Section A.4 of Attachment 5 to the LAR, the licensee states that the PRA logic models for the extended CT configuration did not introduce any new common cause events because no new failure modes were added that required a CCF assessment. In Table A.10.1, Item 26 of Attachment 5, CCF events involving the ESCWS chillers are not a dominant contributor to risk. The PRA analysis for the LAR includes CCF of other SSCs that are included in the PRA. The NRC staff has reviewed the licensee's assessment for the potential introduction of new CCF mechanisms. The NRC staff finds that the licensee has adequately assessed that the proposed changes do not degrade defenses against potential common-cause failures and directly considers the impact of the common-cause initiator.

3.2.5 Maintain multiple fission product barriers

In Section 3.5 (Item 5) of Attachment 1 to the LAR, the licensee states that the proposed ESCWS CT change does not directly impact any of the three fission product barriers (fuel cladding, reactor coolant system, containment building) or otherwise cause their degradation.

The NRC staff has reviewed the licensee's assessment and notes that the proposed changes do not significantly increase the likelihood or consequence of an event that challenges multiple fission product barriers, and does not introduce a new event that would challenge these barriers. The NRC staff finds that the proposed changes do not affect the independence of the fission product barriers, and therefore, the independence of those barriers is not degraded.

3.2.6 Preserve sufficient defense against human errors

In Section 3.5 (Item 6) of Attachment 1 to the LAR, the licensee states that they will provide oversight and support for emergent issues in the event operator actions are required. Defense-in-depth measures include operator actions for the CSIP area cooling, Auxiliary Relay Cabinet Room cooling, and Switchgear Room cooling following a loss of HVAC to these areas. Prior to exceeding 72 hours from the time of TS 3.7.13 LCO entry, these actions will be briefed by the licensee with Operations. The fan used for the CSIP area cooling will be pre-staged and verified to be functional at this time also. Additionally, prior to exceeding 72 hours from the time of TS 3.7.13 LCO entry, equipment will be protected, and administrative controls will be in place to support the compensatory measures. Pre-job briefs will be conducted by the licensee prior to and during the evolution to reinforce good human performance behaviors and barriers that reduce risk. The opposite train of all associated TS will be protected during the CT. Duke Energy fleet staff will be available to support Harris plant staff with resolution of issues during the proposed CT. The NRC staff finds that the proposed changes preserve defenses against human error, because no new human error mechanisms are introduced.

3.2.7 Continue to meet the intent of the plant's design criteria

In Section 3.5 (Item 7) of Attachment 1 to the LAR, the licensee states that the proposed changes do not modify the plant design, or the design criteria applied to SSCs during the licensing process. Compensatory measures, as described in Section 3.6 and Attachment A.9 of the LAR, will be implemented prior to exceeding 72 hours from the time of TS 3.7.13 LCO entry. The NRC staff finds that the intent of the plant's design criteria is maintained by the proposed changes.

3.3 Principle 3: Evaluation of Safety Margins

Regulatory Position C.2.1.2 in RG 1.174 discusses two specific criteria that should be addressed when considering the impact of the proposed changes on safety margin:

- Codes and standards or their alternatives approved for use by the NRC are met, and
- Safety analyses acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or the proposed revisions provide sufficient margin to account for uncertainty in the analysis and data.

The design and operation of the ESCWS is not modified by this LAR. Increasing the CT by 96 hours does not add or remove anything from the ESCWS as designed. The design heat removal capacity remains unchanged during the CT, assuming the alternate ESCWS train continues to operate as designed. No codes or standards approved for use by the NRC relevant to the ESCWS and associated systems are modified or affected. No TS safety limits are affected. The LAR states that no safety analysis acceptance criteria stated in the FSAR are impacted by the amendment.

Although the licensee will be able to have the ESCWS equipment out of service longer than the current TS allows and the likelihood of successful fulfillment of the ESCWS function will be decreased when redundant train(s) is not available, the NRC staff finds that the capability to fulfill the function will be retained when the available equipment functions as designed. Any increase in ESCWS unavailability because associated equipment can be unavailable for a longer time is evaluated in Section 3.4 below, which shows margin to the acceptance guidelines in RG 1.174 and RG 1.177. The NRC staff finds that safety margins are not affected adversely by the extension of the CT and adequate margin of safety will continue to be maintained by the proposed change.

3.4 Principle 4: Change in Risk Consistent with the Commission's Policy Statement on Safety Goals

RG 1.177 addresses Principle 4 through a three-tiered approach for evaluating risk associated with proposed changes to TS CTs:

- Tier 1 assesses the risk impact of proposed TS CT changes in accordance with acceptance guidelines in RG 1.174 and RG 1.177, consistent with the Commission's policy statement on safety goals for the operations of nuclear power plants. The risk impact is evaluated against: (1) operational plant risk as represented by the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF); and (2) incremental plant risk while equipment covered by the proposed TS CT changes are out-of-service, as represented by the ICCDP and the ICLERP. The Tier 1 evaluation also addresses acceptability of the plant-specific PRAs used to assess the changes in risk.

- Tier 2 identifies and evaluates any potential risk-significant plant configurations that could result if any equipment, in addition to that associated with the proposed TS CT changes, will be taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are involved. The purpose of this evaluation is to ensure that there are appropriate restrictions on dominant risk-significant equipment configurations associated with the proposed TS CT changes. In addition, compensatory measures that can mitigate any corresponding increase in risk are identified and evaluated.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures have been established for identifying risk-significant plant configurations resulting from maintenance or other operational activities, and that appropriate compensatory measures are taken to avoid risk-significant configurations that may not have been considered in the Tier 2 evaluation.

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making. For proposed changes resulting in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals.

3.4.1 Tier 1 Evaluation – Risk Impact

In accordance with Tier 1 outlined in RG 1.177, the licensee should assess the impact of the proposed TS CT changes as represented by the ΔCDF, ICCDP, ΔLERF, and ICLERP. As part of this evaluation, the licensee should demonstrate that its PRA (or its qualitative analyses, bounding analyses, detailed analyses, or compensatory measures if a PRA of sufficient scope is

not available) is acceptable for assessing the proposed TS CT changes. Also, uncertainties should be appropriately considered in the analyses and interpretation of findings. The Tier 1 review involves two aspects: (1) evaluation of the technical acceptability of the Harris PRAs used to support this application, and (2) evaluation of the PRA results and insights based on the licensee's proposed changes. In addition, the NRC staff notes that the total CDF and total LERF estimated for the plant, and the ΔCDF and ΔLERF estimated for the TS change should be consistent with the risk acceptance guidelines in RG 1.174. The sections that follow present the NRC staff's assessment of the LAR, as supplemented, regarding:

- PRA acceptability (SE Section 3.4.1.1),
- PRA results and insights (SE Section 3.4.1.2), and
- PRA sensitivity and uncertainty analyses (SE Section 3.4.1.3).

3.4.1.1 PRA Acceptability

Section C.2.3. of RG 1.174 states, "[t]he PRA analysis used to support an application is measured in terms of its appropriateness with respect to scope, level of detail, conformance with the technical elements, and plant representation. These aspects of the PRA are to be commensurate with its intended use and the role the PRA results play in the integrated decision process." The acceptability of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the acceptability of the PRA. This applies to Tier 1, and it also applies to Tier 2 and Tier 3 to the extent that a PRA model is used.

The sections that follow present the NRC staff's assessment of acceptability of the licensee's PRA (i.e., internal events, internal flooding, high winds, and internal fire PRAs), quantitative seismic analysis, and qualitative analyses of other external hazards relative to the four aspects of PRA:

- Scope of PRA (SE Section 3.4.1.1.1),
- Conformance of PRA with the technical elements, and acceptability of other external hazard analyses (SE Section 3.4.1.1.2),
- Level of detail in PRA (SE Section 3.4.1.1.3), and
- Plant representation in PRA (SE Section 3.4.1.1.4).

3.4.1.1.1 Scope of the PRA

Regulatory Position C.2.3.2 of RG 1.177 states that the licensee should perform evaluations of CDF and LERF to support any risk-informed changes to TS. The scope of the analysis should include all hazard groups (i.e., internal events, internal flooding, fires, seismic events, high winds, and other external hazards) unless it can be shown the contribution from specific hazard groups does not affect the decision. In some cases, a PRA of sufficient scope may not be available. This will have to be compensated for by qualitative arguments, bounding analyses, or compensatory measures.

Based on the LAR, as supplemented, the change in risk (i.e., ΔCDF, ΔLERF, ICCDP, and ICLERP) resulting from the proposed ESCWS CT extension is estimated utilizing PRAs for at-power internal events, internal flooding, internal fire, and high winds. Other external hazards, qualitative assessments were used to screen these events from further consideration. Based on its review of the LAR, as supplemented, the NRC staff finds that the licensee's risk assessment, when compared to the regulatory positions contained in RGs 1.174 and 1.177, is of sufficient scope for use in this specific risk-informed application.

3.4.1.1.2 Conformance of PRA with the Technical Elements, and Acceptability of Other External Hazard Analyses

RG 1.200 describes one acceptable approach for determining whether the acceptability of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision-making for light-water reactors. RG 1.200 endorses, with clarifications and qualifications, American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS RA-Sa-2009 (Reference 12), "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." The ASME/ANS PRA standard provides technical supporting requirements (SRs) in terms of three Capability Categories. The intent of the delineation of the Capability Categories within the SRs is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from Capability Category I to Capability Category III. Per RG 1.200, Capability Category II of the ASME/ANS standard, is the level of detail that is adequate for the majority of applications.

The licensee used its internal events, internal flooding, internal fire, and high winds PRAs to support this application. The following present NRC staff's assessment of these PRAs and their conformance with the technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, for use in supporting this risk-informed application. Also, the NRC staff's assessment of other external hazards is discussed.

Internal Events PRA

The NRC staff review of the internal events PRA is based on: (1) the results of the peer review of the internal events PRA and the associated facts and observations (F&Os) closure review described in Section A.2.1 of Attachment 5 to the LAR; and (2) the previously docketed information relevant to the NRC staff's review of the internal events PRA for Harris' adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors" (Reference 13), and for relocation of surveillance frequency requirements to a licensee control program, Technical Specifications Task Force (TSTF) Traveler TSTF-425 (Reference 14).

The internal events PRA was subject to a full-scope peer review in 2002 in accordance with the guidance in Nuclear Energy Institute (NEI) 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance" (Reference 15), and self-assessments in accordance with Appendix B of RG 1.200 to address aspects of the PRA technical SRs not considered in the 2002 peer review. In March 2017, an F&Os closure review was performed by an independent assessment team on all internal events finding-level F&Os. Based on the NRC staff's safety evaluation for Harris' adoption of 10 CFR 50.69, the F&Os closure review was performed in accordance with Appendix X (Reference 16) to the guidance in NEI 05-04 (Reference 17), NEI 07-12 (Reference 18), and NEI 12-13 (Reference 19), concerning the process "Close-Out of

Facts and Observations," as accepted by the NRC in a letter dated May 3, 2017 (Reference 20). The March 2017 closure review closed out all open internal events F&Os.

In Section A.2.1 of Attachment 5 to the LAR, the licensee describes the 2017 and 2018 internal events PRA model updates, including the incorporation of Diverse and Flexible Mitigation Strategies (FLEX) equipment to mitigate accident sequences. The PRA updates performed by the licensee were not considered a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200. Item 15 of LAR Table A.10.1 stated that the accident sequences involving FLEX do not involve ESCWS, and a sensitivity analysis that removed FLEX from the PRA model indicates that the FLEX has a very small contribution in total delta risk. Therefore, the modeling of FLEX in the PRA does not impact the conclusions of this LAR.

In addition, the LAR (Section 3.2 of Attachment 1, and Tables A.10.1 and A.10.2 of Attachment 5), as supplemented, discussed the results of a room heatup analysis that determined the expected area temperatures in the RAB when the ESCWS system is not functioning. The heatup analysis determined that most areas analyzed will not exceed unacceptable temperature values within 24 hours following a loss of ESCWS. The Harris PRA only models the chillers for the Charging/Safety Injection Pump Rooms and the Switchgear Rooms in accordance with the room heatup analysis. The NRC staff has reviewed the licensee's analysis and finds that the licensee has followed the guidance in RG 1.200 for determining the technical acceptability of the internal events PRA. Conformance of the internal events PRA to the applicable technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, is acceptable to the extent needed to support this application.

Internal Flooding PRA

In Section A.2.2 of Attachment 5 to the LAR, the licensee indicated that the Harris internal flooding PRA model was upgraded in 2014 to meet the requirements of the ASME PRA standard and RG 1.200, Revision 2. A focused-scope peer review of the internal flooding PRA was conducted by the licensee following the guidance of NEI 05-04 to assess the model against the SRs of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200. In March 2017, an F&Os closure review was performed by an independent assessment team on all open internal flooding finding-level F&Os. Based on the NRC staff's safety evaluation for Harris' adoption of 10 CFR 50.69, the F&Os closure review was performed in accordance with Appendix X to the guidance in NEI 05-04, NEI 07-12, and NEI 12-13, as accepted by the NRC in a letter dated May 3, 2017. During the Harris F&Os closure review, fifteen of twenty-five internal flooding finding-level F&Os were determined to have been adequately addressed and were closed.

In Section B.2 of Attachment 5 to the LAR, the licensee lists the ten F&Os that remained open after the F&Os closure review. The licensee provided a disposition for each open F&O as it relates to this application. The NRC staff evaluated each open F&O and the licensee's disposition to determine whether the F&O had any significant impact on the application. The NRC staff finds the open internal flooding F&Os were properly assessed and dispositioned to support the proposed TS CT change. The NRC staff has reviewed the licensee's analysis and finds that the licensee has followed the guidance in RG 1.200 for determining the technical acceptability of the internal flooding PRA. Conformance of the internal flooding PRA to the applicable technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, is acceptable to the extent needed to support this application.

Internal Fire PRA

The NRC staff review of the internal fire PRA is based on: (1) the results of the peer review of the internal fire PRA and the associated F&Os closure review described in Section A.2.3 of Attachment 5 to the LAR; and (2) the previously docketed information relevant to the NRC staff's review of the internal fire PRA for Harris' adoption of 10 CFR 50.69, and for relocation of surveillance frequencies to licensee control, TSTF-425.

The internal fire PRA was subject to both an NRC review and full-scope industry peer review in 2008 during the review associated with approval of Harris' license amendment to implement 10 CFR 50.48(c) (Reference 21). The 2008 review was conducted using the ANSI/ANS-58.23-2007 standard, whereas RG 1.200, Revision 2, endorses ASME/ANS RA-Sa 2009, with clarifications and qualifications. The ASME/ANS-RA-Sa-2009 standard states that it was assembled from the ANSI/ANS-58.23-2007 fire PRA standard. In Table B.3 of Attachment 5 to the LAR, the licensee assessed the differences between ANSI/ANS-58.23-2007 and the current version of the fire PRA standard in ASME/ANS RA-Sa-2009 and confirmed there were no technical differences between the two versions of the standard. In October 2017, an F&Os closure review was performed by an independent assessment team on fire PRA finding-level F&Os in accordance with Appendix X to the guidance in NEI 05-04, NEI 07-12, and NEI 12-13, as accepted by the NRC in a letter dated May 3, 2017. Six finding-level F&Os remain open after the F&Os closure review.

In Section B.3 of Attachment 5 to the LAR, the licensee provides the internal fire PRA finding-level F&Os that remain open and the licensee's disposition to these F&Os for this LAR. Each F&O was dispositioned by either providing a description of how the F&O was resolved or providing an assessment of the impact of the F&O resolution on the LAR results. The NRC staff evaluated each F&O and the licensee's disposition to determine whether the F&O had any significant impact on the application. The NRC staff finds the internal fire PRA F&Os were properly assessed and dispositioned to support the proposed TS CT change.

Additionally, the licensee identified that SRs FSS-D7 and FSS-D9 of ASME/ANS-RA-Sa-2009 were met at Capability Category I but stated that there were no F&Os related to these SRs. Based on the safety evaluation associated with approval of Harris' license amendment to implement 10 CFR 50.48(c), the NRC staff finds that meeting Capability Category I (as opposed to Capability Category II) for these SRs has a minor impact on the risk results for this LAR and, therefore, would not change the conclusions of this LAR.

As discussed previously, the 2017 and 2018 internal events PRA model updates are not considered an PRA upgrade. In the supplement dated September 3, 2019, the licensee stated that the internal fire PRA supporting this application does not incorporate all the 2017 and 2018 updates to the internal events PRA. However, the incorporation of all the updates resulted in a lower fire ICCDP, ICLERP, ΔCDF, and ΔLERF for the assessed CT compared to the licensee's initial submittal of this LAR. Therefore, the initial assessment, without incorporation of the 2017 and 2018 internal events updates, bounds the risk impact with the updates incorporated, and the updates can be excluded for consideration in the context of this application.

The NRC staff has reviewed the licensee's analysis and finds that the licensee has followed the guidance in RG 1.200 for determining the technical acceptability of the internal fire PRA. Conformance of the internal fire PRA to the applicable technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, is acceptable to the extent needed to predict the change in CDF and LERF for use in this LAR.

High Winds PRA

In Attachment 5 of the LAR, the licensee indicated that Harris' high winds PRA was peer reviewed in 2015 against the requirements of RG 1.200, Revision 2. Four finding-level F&Os were generated from the peer review, and Section B.4 of Attachment 5 to the LAR, as supplemented by the response to RAI 02 in letter dated September 3, 2019, dispositions these F&Os. The NRC staff evaluated each open F&O and the licensee's disposition to determine whether the F&O had any significant impact on the application. The NRC staff finds the open high winds F&Os were properly assessed and do not impact the TS change requested in this LAR.

In the supplement dated September 3, 2019, the licensee stated that the high winds PRA supporting this application does not incorporate all the 2017 and 2018 updates to the internal events PRA. However, the licensee provided a justification that high winds are not considered a significant hazard for Harris and have a negligible contribution to risk for this application. The NRC staff reviewed this justification and finds the licensee has appropriately evaluated high winds to the extent needed to support this application in accordance with RG 1.177 and determined this hazard does not impact this application. Therefore, the high wind PRA results provided in the initial submittal in the calculation of total and delta risk for this application are considered by the NRC staff to be conservative values. The NRC staff has reviewed the licensee's analysis and conformance of the high winds PRA to the applicable technical elements in the ASME/ANS PRA standard endorsed by RG 1.200, is acceptable to the extent needed to support this application.

External Flooding Impact

In Section A.5.5 of Attachment 5 to the LAR, the licensee provides a qualitative evaluation for the external flooding hazard and concludes that this hazard has no impact on risk for this application. The licensee states that Harris completed a Flood Hazard Reevaluation Report, and its results indicate some flood levels exceed the current licensing basis (CLB) flood levels. However, the increased levels do not exceed the flood protection capabilities. Therefore, taking a train of the ESCWS out of service is assessed to be unaffected by an external flooding during the extended CT. The NRC staff finds that the licensee provides adequate justification to screen out the external flooding impact was provided. The NRC staff concludes that the external flooding impact is relatively low to this application, because the flood levels do not exceed the plant flood protection capabilities.

Seismic Risk

In Section A.5.6 of Attachment 5 to the LAR, the licensee provides a qualitative evaluation for seismic risk using a comparison of the safe shutdown earthquake (SSE) and ground motion response spectra (GMRS) based on the NRC's "Support Document for Screening and Prioritization Results Regarding Seismic Hazard Re-Evaluations for Operating Reactors in the Central and Eastern United States" (Reference 22). Based on the comparison, the licensee concludes that the risk from seismically induced failure of the ESCWS is a very low probability event. In Attachment 4 of the Harris 50.69 LAR (Reference 23), the licensee uses the Seismic Margin Assessment developed for Individual Plant Evaluation-External Events for categorization. The NRC staff notes that the cited document provides site-specific seismic hazard evaluation, but not the risk from seismic events.

For the currently operating plants, the SSE was developed to envelope the deterministic hazard at the site. Therefore, from a seismic hazard perspective, the site-specific SSE, derived using a deterministic approach, can be compared to the corresponding GMRS, which is derived using a probabilistic approach. The comparison of the site-specific GMRS and SSE provides information about any seismic risk that would be unaccounted for in the current plant licensing basis. The comparison does not provide an estimate of the seismic risk or its impact on this application.

As part of its evaluation, the NRC staff used the plant-specific re-evaluated seismic hazard curves developed in response to the Near-Term Task Force recommendation 2.1 (Reference 24), and a plant level high confidence of low probability of failure capacity of 0.29g peak ground acceleration as well as the composite beta factor of 0.4 used for the Harris in the Generic Issue (GI)-199 (Reference 25) to determine the impact of seismic risk on the proposed change. Based on its review and evaluation, the NRC staff determined that consideration of seismic risk will not change the conclusions of this LAR, because consideration of seismic risk adds a small contribution to the calculated change in risk for this application and margin is available that meets the acceptance guidelines for this TS CT LAR.

Other External Event Hazards

The LAR, as supplemented, does not evaluate other external hazards to determine whether they impact the application. However, the NRC staff finds that previously docketed information in Attachments 4 and 5 of the Harris LAR to adopt 10 CFR 50.69 (Reference 23) provides a qualitative assessment of each of the following external hazards:

- Aircraft Impact
- Avalanche
- Biological Event
- Coastal Erosion
- Drought,
- Fog
- Forest or Range Fire
- Frost
- Hail
- High Summer Temperature
- High Tide, Lake Level or River Stage
- Ice Cover
- Industrial or Military Facility Accident
- Landslide
- Lightning
- Low Lake Level or River Stage
- Low Winter Temperature
- Pipeline Accident
- Meteorite or Satellite Impact
- Release of Chemicals on Onsite Storage
- River Diversion
- Sand or Dust Storm
- Seiche
- Snow
- Soil Shrink-Well Consolidation
- Storm Surge
- Toxic Gas
- Transportation Accident
- Tsunami
- Turbine-Generated Missiles
- Volcanic Activity
- Waves

This list is essentially the same list of hazards as presented in Table 4-1 of NUREG 1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making” (Reference 26). Each of these were screened from further consideration based on the screening criteria in Section 6 of ASME/ANS RA-Sa-2009. The NRC staff has reviewed the licensee’s information in the Harris LAR to adopt 10 CFR 50.69 for all other external hazards. The NRC staff finds the licensee has appropriately evaluated other external hazards to the extent needed to support this application in accordance with RG 1.177 and determined those hazards do not impact this application.

3.4.1.1.3 Level of Detail in PRA

Section C.2.3.3 of RG 1.174 states, the level of detail required of the PRA is that which is sufficient to model the risk impact of the proposed changes. If the impacts of the proposed changes to the plant cannot be associated with elements of the PRA, the PRA should be modified accordingly, or the impact of the change should be evaluated qualitatively as part of the integrated decision-making process. In any case, the licensee should properly account for the effects of the changes on the reliability and unavailability of SSCs or on operator actions.

In Section A.3 of Attachment 5 to the LAR, the licensee provides the general assumptions of the PRA analysis and an evaluation of their impact on this application in LAR Table A.10.6. The assumption regarding the human failure events (HFEs) to open doors and implement portable fans as an alternate means of cooling the switchgear rooms/CSIP rooms is considered a key source of uncertainty and is further evaluated in a sensitivity analysis discussed in SE Section 3.4.1.3. Based on the large margin by which this sensitivity analysis met the RG 1.177 risk acceptance guidelines, the NRC staff concludes that this PRA model uncertainty is not sufficient to change the conclusions of the LAR.

3.4.1.1.4 Plant Representation in PRA

Section C.2.3.4 of RG 1.174 states, the PRA results used to support an application should be derived from a PRA that represents the as-built and as-operated plant to the extent needed to support the application. Consequently, the PRA should have been maintained and updated, where necessary, to ensure it represents the as-built and as-operated plant. In Section A.6 of Attachment 6 to the LAR, the licensee describes the PRA configuration and control program to maintain and update the Harris PRA such that the PRA represents the as-built, as-operated plant. The licensee has procedures to provide the guidance, requirements, and processes for the maintenance, update, and upgrade of the PRA. As part of this program, the licensee evaluates and prioritizes changes in PRA inputs, as well as address discovery of new information that could affect the PRA. Any identifiable plant change is analyzed for its risk significance.

All plant changes not yet incorporated into the PRA (i.e., open items) are tracked and reviewed prior to the start of an application for their impact on that application. The licensee provides in Section A.6 of Attachment 6 to the LAR the disposition of three open items that have a medium risk impact and concludes that they have no impact on the proposed TS CT change application. The NRC staff has reviewed the licensee’s open items and associated dispositions and concludes that the open items have a negligible impact on this application. Based on the licensee's PRA configuration and control program to maintain and update the PRA, the NRC staff finds the PRA results used to support this application are derived from an integrated PRA that represents the as-built and as-operated plant to the extent needed to support the application.

3.4.1.2 PRA Results and Insights

Section C.2.5.2 of RG 1.174 indicates that the mean values of the risk metrics (i.e., CDF, LERF, ICCDP, ICLERP, ΔCDF, and ΔLERF) should be compared against the applicable risk acceptance guidelines. The mean values referred to are the means of the risk metric’s probability distributions that result from the propagation of the uncertainties on the PRA input parameters.

The licensee evaluated the risk impact of the proposed changes using Harris' PRA models for internal events, internal flooding, internal fires, and high winds, provided in Sections A.5.1 to A.5.4 of Attachment 5 to the LAR. Table A.5.8 of Attachment 5 to the LAR provides these risk results for the proposed TS CT extension.

The licensee calculated a total CDF and LERF of 3.48×10^{-5} per year and 6.53×10^{-6} per year, respectively, based on the CT configuration case. The licensee calculated a ΔCDF and ΔLERF for the proposed TS CT extension as follows (numbers are rounded for ease of use):

ΔCDF = 9×10^{-7} /year
(RG 1.174 Acceptance Guideline: 1×10^{-5} /year, Region II in Figure 4 of RG 1.174)

ΔLERF = 2×10^{-8} /year
(RG 1.174 Acceptance Guideline: 1×10^{-6} /year, Region II in Figure 5 of RG 1.174)

The licensee calculated the ICCDP and ICLERP for the proposed TS CT extension as follows:

ICCDP = 2×10^{-8} (RG 1.177 Acceptance Guideline: 1×10^{-6})

ICLERP = 3×10^{-10} (RG 1.177 Acceptance Guideline: 1×10^{-7})

The risk results provided by the licensee represent point estimate values (obtained by quantification of the cutset probabilities using mean values for each basic event probability) and are not true mean values for these risk metrics. In general, the point estimate values of these risk metrics, obtained by quantification of the cutset probabilities using mean values for each basic event probability, does not produce a true mean value for these risk metrics. However, due to the large margin by which the point estimate values meet the RG 1.174 and RG 1.177 risk acceptance guidelines, the NRC staff concludes that use of the corresponding true mean values would not change the conclusions of this assessment. The NRC staff has reviewed the licensee's assessment and finds that the risk increase for the proposed ESCWS CT extension is consistent with RG 1.174, Section C.2.4 and RG 1.177, Section C.2.4.

3.4.1.3 Sensitivity and Uncertainty Analyses

Section C.2.5 of RG 1.174 identifies the following types of uncertainty that affect the results of PRAs: parameter uncertainty, model uncertainty, and completeness uncertainty. In accordance with regulatory positions in RGs 1.174 and 1.177, uncertainties should be appropriately considered in the analysis and interpretation of findings. Also, RG 1.174 states, the results of the sensitivity studies should confirm the guidelines are still met even under the alternative assumptions.

In Section A.10 of Attachment 5 to the LAR, the licensee addresses the sources of modeling uncertainty. Using Electric Power Research Institute (EPRI) Report 1016737 (Reference 27), as a guideline, the licensee provides in Table A.10.1 of the LAR the detailed generic uncertainty evaluations in the areas of initiating event analysis, accident sequence analysis, success criteria, system analysis, human reliability analysis, internal flooding, and LERF analysis with applicability and resolution for this LAR application. In addition, the licensee provides the evaluations of the internal events model uncertainty in Table A.10.2 of the LAR, internal fire model uncertainty in Table A.10.3 of the LAR, high winds model uncertainty in Table A.10.4 of the LAR, and internal flooding model uncertainty in Table A.10.5 of the LAR. The licensee

provides the general assumptions to the PRA analysis in Section A.3 of Attachment 5 to the LAR, and an evaluation of their impact on this application in Table A.10.6 of the LAR.

From the uncertainty analyses described above, the licensee identified one key source of model uncertainty associated with the HFEs to open doors and implement portable fans as an alternate means of cooling the switchgear rooms/CSIP rooms. In Section A.10 of Attachment 5 to the LAR, as supplemented, the licensee performed a sensitivity analysis assuming guaranteed failure of operator actions to provide an alternate means of cooling the switchgear rooms/CSIP rooms. The results of this sensitivity analysis (i.e., ICCDP, ICLERP, ∆CDF, ∆LERF) still meet the RG 1.174 and RG 1.177 risk acceptance guidelines by a large margin. This provides additional confidence that any uncertainties associated with this operator action would not change the conclusions of this assessment. The NRC staff finds that the licensee followed an acceptable process to identify key assumptions, dispositioned key assumptions for this application, and performed its sensitivity and uncertainty analyses consistent with RG 1.174. The results are robust considering the uncertainty, and, therefore, are acceptable to the extent needed to support this application.

3.4.2 Tier 2 Evaluation – Avoidance of Risk-Significant Plant Configurations

Section C.2.3 of RG 1.177 discusses Tier 2 of the three-tiered approach for evaluating risk associated with proposed changes to TS CT. According to Tier 2, the avoidance of risk-significant plant configurations limits potentially high-risk configurations that could exist if equipment, in addition to that associated with the proposed changes, are simultaneously removed from service or other risk-significant operational factors, such as concurrent system or equipment testing, are involved. Therefore, a licensee's Tier 2 evaluation should identify the dominant risk-significant configurations relevant to the proposed TS CT change and ensure appropriate restrictions are placed on these configurations (e.g., assess whether certain enhancements to the TS or procedures are needed to avoid these plant configurations). In addition, compensatory measures that can mitigate any corresponding increase in risk should be identified and evaluated.

In Section A.7 of Attachment 5 to the LAR, the licensee provided configuration-specific insights. The licensee performed analyses to identify risk-significant combinations of equipment out-of-service during the extended CT and identified further compensatory actions, as discussed in Section 3.2.7 of this SE and added in the TS Bases of 3/4.7.13, to avoid these risk-significant configurations during the extended CT. In addition, in Table A.7 of Attachment 5 to the LAR, the licensee provided a list of SSCs whose unavailability should be minimized during the CT, based upon Fussell-Vesely, Risk Achievement Worth, and Risk Reduction Worth importance measures. The NRC finds that the licensee provided adequate analyses of risk-significant configurations during the extended CT and identified appropriate compensatory actions that can mitigate corresponding increases in risk. Based on these findings, the NRC staff concludes that the licensee's analysis of risk-significant plant equipment outage configurations will be avoided during the extended CT and is consistent with the guidance of RG 1.177.

3.4.3 Tier 3 Evaluation – Risk-Informed Configuration Risk Management

Section C.2.3 of RG 1.177 discusses Tier 3 of the three-tiered approach for evaluating risk associated with proposed changes to TS CT. Tier 3 is the establishment of an overall CRMP to ensure other potentially lower probability, but nonetheless risk-significant, configurations resulting from maintenance and other operational activities are identified and managed. Because the Maintenance Rule, as codified in 10 CFR 50.65(a)(4), requires licensees to assess

and manage the potential increase in risk that may result from activities such as surveillance testing, and corrective and preventive maintenance, a licensee may use its existing Maintenance Rule program to satisfy Tier 3.

In Section A.8 of Attachment 5 to the LAR, the licensee stated that Harris has an established configuration risk management program that implements 10 CFR 50.65(a)(4) requirements. Harris uses the Equipment Out of Service software tool to analyze plant risk in both real time and a look-ahead of plant configurations over specified period of time. During the CT, risk will be monitored in real time and any emergent risk configurations will be addressed appropriately. The licensee's configuration risk management program requires the implementation of risk management actions to help alleviate risk when unforeseen events put the plant in a risk significant configuration. Thus, plant risk will be effectively managed prior to and during the extended CT. The NRC staff has reviewed the licensee's assessment and finds that the licensee's Tier 3 program is consistent with the guidance in RG 1.177 and finds the proposed change acceptable.

3.5 Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the changes to the TS. An implementation and monitoring program intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change.

RG 1.177 states that the licensee is to use a three-tiered approach in implementing the proposed TS CT changes. Application of the three-tiered approach is in keeping with the fundamental principle that the proposed changes are consistent with the defense-in-depth philosophy. Application of the three-tiered approach provides assurance that defense-in-depth will not be significantly impacted by the proposed changes. Furthermore, RG 1.177 states that, to ensure that extension of a TS CT does not degrade operational safety over time, the licensee should ensure, as part of its Maintenance Rule program (10 CFR 50.65), that when equipment does not meet its performance criteria, the evaluation required under the Maintenance Rule includes prior related TS changes in its scope.

The licensee provides an evaluation of the proposed TS change against the three-tiered approach in Attachment 5 to the LAR. In Section A.9 of Attachment 5 to the LAR the licensee proposes five compensatory actions that can mitigate any corresponding increase in risk associated with the proposed changes. In addition, the ESCWS is monitored under the Harris Maintenance Rule program. If the established Maintenance Rule program reliability or availability performance criteria for the ESCWS is exceeded, they are evaluated for 10 CFR 50.65(a)(1) actions, which requires increased management attention and goal setting in order to restore their performance to an acceptable level.

The NRC staff reviewed the proposed deletion of the expired footnotes and determined the deletions are acceptable because the footnotes were temporary and can no longer be applied to the associated TS. The NRC staff reviewed the proposed new footnotes which represent a risk-informed LAR. One acceptable approach for making risk-informed decisions about proposed TS changes is to show that the proposed changes meet the five key principles stated in RG 1.177. The NRC staff also finds that the implementation and monitoring program for the proposed TS change described by the licensee satisfies the fifth key principle of RG 1.174 and RG 1.177.

3.6 NRC Staff Conclusion

The NRC staff reviewed the proposed changes to permanently modify the Harris TSs to permit one train of the ESCWS to be inoperable for up to 7 days from the currently allowed 72 hours for extended maintenance activities on the ESCWS and air handlers supported by the ESCWS for equipment reliability. The amendment would also remove an expired Note in numerous TS sections that was previously added by implementation of License Amendment No. 153.

Based on its review of the Harris ESCWS CT LAR, as supplemented, the NRC staff concludes the Harris PRA (i.e., internal events, internal flooding, high winds, and internal fire PRAs) and non-PRA analysis are acceptable for assessing risk to the extent needed to support this application. The NRC staff based this conclusion on the findings that, for this risk-informed application and to the extent needed to support the application: (1) the licensee's risk assessment is of sufficient scope; (2) the Harris internal events, internal flooding, internal fire, and high winds PRAs appropriately conform to the applicable technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, to the extent needed to predict the ΔCDF and ΔLERF, (3) the risk from external flooding, seismic events, and other external hazards not addressed using PRA do not impact this application; (4) the level of detail in the PRA models and the PRA assumptions are appropriate to evaluate the risk impact for this application; and (5) the PRAs represents the as-built and as-operated plant.

Based on these findings, the NRC staff determined the compensatory measures are acceptable and concludes that the licensee's request to permanently revise the ESCWS TS CT from 72 hours to 7 days follows the three-tiered approach and performance monitoring programs outlined in RG 1.177 and meets the five key principles outlined in RG 1.174. The NRC staff finds that the proposed changes do not significantly affect the seven considerations for defense-in-depth and the proposed changes preserve defense-in-depth commensurate with the expected frequency and consequence of challenges to the system resulting from the proposed changes. The NRC staff concludes that there is reasonable assurance that the proposed TS changes will have minimal impact on the licensee's ability to continue to comply with the requirements of 10 CFR 50.36, GDC 2, GDC 4, GDC 44, GDC 45, and GDC 46.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment on February 18, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (84 FR 31632). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

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Date: March 31, 2020

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 176 REGARDING THE EXTENSION OF THE ESSENTIAL SERVICES CHILLED WATER SYSTEM ALLOWED OUTAGE TIME AND REMOVAL OF AN EXPIRED NOTE FROM TECHNICAL SPECIFICATIONS (EPID L-2019-LLA-0025) DATED MARCH 31, 2020

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ADAMS Accession No.: ML20050D371 *by memorandum **by e-mail

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