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

June 14, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 342 RE: ONE TIME EXTENSION TO TECHNICAL SPECIFICATIONS 3.5.1, 3.6.1.9, AND 3.6.4.1 COMPLETION TIMES TO SUPPORT RESIDUAL HEAT REMOVAL PUMP MOTOR REPLACEMENT **(EMERGENCY CIRCUMSTANCES)**(EPID L-2021-LLA-0110)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 342 to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the technical specifications (TS) in response to your application dated June 12, 2021, as supplemented by two letters dated June 13, 2021, and a letter dated June 14, 2021, which superseded all previous letters.

The amendment modifies TS 3.5.1, ECCS [Emergency Core Cooling System] – Operating Condition A, TS 3.6.4.1, Secondary Containment, Condition A, and TS 3.6.1.9 RHR [Residual Heat Removal] Containment Spray System, as well as certain Surveillance Requirements to support emergent repair of the "A" residual heat removal pump motor. Specifically, the amendment revises the completion time from 7 days to 34 days for the "A" residual heat removal pump, the completion time from 4 hours to 30 hours for restoring secondary containment, and the completion time from 7 days to 27 days for restoring one containment spray subsystem to operable status. Additionally, the amendment allows extending the completion of several surveillance requirements of equipment being protected during the replacement of the "A" residual heat removal pump motor.

The license amendment is issued under emergency circumstances as provided in the provisions of paragraph 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* due to the time-critical nature of the amendment. In this instance, an emergency situation exists in that the amendment is needed to allow the licensee to avoid a plant shutdown.

A copy of our related Safety Evaluation is also enclosed. The Safety Evaluation describes the emergency circumstances under which the amendment is issued and the final no significant hazards determination. Notice of Issuance addressing the final no significant hazards determination and opportunity for a hearing associated with the emergency circumstances will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Justin C. Poole, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Amendment No. 342 to DPR-59
2. Safety Evaluation

cc: Listserv

EXELON FITZPATRICK, LLC

AND

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 342
Renewed Facility Operating License No. DPR-59

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

   A. The application for amendment by Exelon FitzPatrick, LLC and Exelon Generation Company, LLC (collectively, the licensees) dated June 12, 2021, as supplemented by two letters on June 13, 2021, and a letter on June 14, 2021, 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. DPR-59 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 342, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2021.06.14 19:38:40 -04'00'

James G. Danna, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: June 14, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 342

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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

Remove Page	Insert Page
Page 3	Page 3

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

Remove Pages	Insert Pages
3.3.3.2-2	3.3.3.2-2
3.3.6.1-4	3.3.6.1-4
3.3.6.1-5	3.3.6.1-5
3.5.1-1	3.5.1-1
3.5.1-3	3.5.1-3
3.5.1-4	3.5.1-4
3.5.3-2	3.5.3-2
3.6.1.3-8	3.6.1.3-8
3.6.1.9-1	3.6.1.9-1
3.6.4.1-1	3.6.4.1-1
3.7.2-4	3.7.2-4
3.7.4-3	3.7.4-3
3.8.1-5	3.8.1-5
3.8.1-6	3.8.1-6
3.8.3-2	3.8.3-2
3.8.3-3	3.8.3-3
3.8.4-3	3.8.4-3
3.8.5-2	3.8.5-2
3.8.6-2	3.8.6-2
3.8.6-3	3.8.6-3

(4) Exelon Generation Company pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools.

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 342, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No. 2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989), 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994), and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985,

SURVEILLANCE REQUIREMENTS

--NOTE--

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours

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SURVEILLANCE		FREQUENCY
SR 3.3.3.2.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	In accordance with the Surveillance Frequency Control Program*
SR 3.3.3.2.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.4	Perform CHANEL CALIBRATION for each required instrumentation channel.	In accordance with the Surveillance Frequency Control Program

* This Surveillance for ST-43I is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS

---------------------------------------NOTES---------------------------------------

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 2.d, 2.g, 7.a, and 7.b; and (b) for up to 6 hours for Functions other than 2.d, 2.g, 7.a, and 7.b provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program*
SR 3.3.6.1.3	-----------------------NOTE---------------------- For Functions 1.f and 2.f, radiation detectors are excluded. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

(continued)

* This Surveillance for ISP-150B is not required to be performed for functions 4.a, 4.b, 4.d, 4.e and 4.f of table 3.3.6.1-1 until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.4	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program*
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program*
SR 3.3.6.1.6	Calibrate the radiation detectors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.7	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.8	-----NOTE----- “n” equals 2 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. ----- Verify the ISOLATION INSTRUMENTATIN RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program

* This Surveillance for ISP-150B is not required to be performed for functions 4.a, 4.b, 4.d, 4.e and 4.f of table 3.3.6.1-1 until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), REACTOR PRESSURE VESSEL (RPV) WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS-Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

-----------------------------------NOTE-------------------------------
Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) cut in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

---NOTE---------------------------------------
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days*
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in Mode 4.	12 hours 36 hours

(continued)

* The Completion Time to return the “A” RHR pump to OPERALE is extended to 34 days, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration the “A” RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition C, D, E, or F not met. <u>OR</u> Two or more required ADS valves inoperable.	G.1 Be in Mode 3. <u>AND</u> G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. <u>OR</u> HPCI System and one or more required ADS valves inoperable.	B.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program*

(continued)

* This Surveillance for ST-4B is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program*
SR 3.5.1.3	Verify ADS pneumatic supply header pressure is > 95 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify the RHR System cross tie valves are closed and power is removed from the electrical valve operator.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.5	Cycle open and closed each LPCI motor operated valve independent power supply battery charger AC input breaker and verify each LPCI inverter output voltage is > 576 V and < 624 V while supplying the respective bus.	In accordance with the Surveillance Frequency Control Program

(continued)

* This Surveillance for ST-4B is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program*
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program*
SR 3.5.3.3	----------------------NOTE--------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1040 psig and ≥ 970 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program*
SR 3.5.3.4	----------------------NOTE--------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program*

* This Surveillance for ST-24J is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	-----NOTE----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ---------- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed or otherwise secured and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program*
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program

* This Surveillance for ST-24J is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

3.6 CONTAINMENT SYSTEMS

3.6.1.9 Residual Heat Removal (RHR) Containment Spray System

LCO 3.6.1.9 Two RHR containment spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR containment spray subsystem inoperable.	A.1 Restore RHR containment spray subsystem to OPERABLE status.	7 days*
B. Two RHR containment spray subsystems inoperable.	B.1 Restore one RHR containment spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

* The Completion Time is extended to 27 days, in support of the “A” RHR pump repairs, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration of the “A” RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours*
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment.	C.1 ------NOTE------ LCO 3.0.3 is not applicable. ------------------ Suspend movement of recently irradiated fuel assemblies in the secondary containment.	 Immediately

* The Completion Time is extended to 30 hours, in support of the "A" RHR pump repairs, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration of the "A" RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours. Multiple entries may be necessary to implement compensatory actions, or to address unforeseen circumstances related to the "A" RHR pump motor replacement.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.2.4	Verify each ESW subsystem actuates on an actual or simulated initiation signal.	In accordance with the Surveillance Frequency Control Program*

* This Surveillance for ST-9BA and ST-9BB is not required to be performed until following the return of the “A” RHR pump to OPRABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	In accordance with the Surveillance Frequency Control Program*

* This Surveillance ST-8Q is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.2	----------------------NOTE--------------------- All EDG subsystem starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. --- Verify each EDG subsystem starts from standby conditions, force parallels, and achieves: a. In ≤ 10 seconds, voltage ≥ 3900 V and frequency ≥ 58.8 Hz; and b. Steady state voltage ≥ 3900 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.3	----------------------NOTE--------------------- 1. EDG loading may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one EDG subsystem at a time. 4. This SR shall be preceded by and immediately follow, without shutdown, a successful performance of SR 3.8.1.2. --- Verify each EDG subsystem is paralleled with normal, reserve, or backfeed power and each EDG is loaded and operates for ≥ 60 minutes at a load ≥ 2340 kW and ≤ 2600 kW.	In accordance with the Surveillance Frequency Control Program*

(continued)

* The Surveillances for ST-9BA and ST-9BB is not required to be performed until following the return of the "A" RHR Pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each day tank contains ≥ 327 gal of fuel oil.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.5	Check for and remove accumulated water from each day tank	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.6	Verify that each EDG fuel oil transfer system operates to automatically transfer fuel oil from its storage tank to the associated day tank.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.1.7	----------------------NOTE---------------------- Only required to be met for each offsite circuit that is not energizing its respective 4.16 kV emergency bus. -- Verify automatic and manual transfer of plant power supply from the normal station service transformer to each offsite circuit.	In accordance with the Surveillance Frequency Control Program
SR 3.8.1.8	----------------------NOTE---------------------- If performance with EDG subsystem paralleled with normal, reserve, or backfeed power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition, the power factor shall be maintained as close to the limit as practicable. -- Verify each EDG subsystem rejects a load greater than or equal to its associated single largest post-accident load, and following load rejection, the frequency is ≤ 66.75 Hz.	In accordance with the Surveillance Frequency Control Program

(continued)

* The Surveillances for ST-9AA, ST-9BB, ST-9BA, and ST-9BA is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more EDGs with new fuel oil properties not within limits.	D.1	Restore stored fuel oil properties to within limit.	30 days
E. One or more EDGs with required starting air receiver pressure < 150 psig and ≥ 110 psig.	E.1	Restore required starting air receiver pressure to within limits.	48 hours
F. Requires Action and associated Completion Time of Condition A, B, C, D, or E not met. <u>OR</u> One or more EDGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other then condition A, B, C, D, or E.	F.1	Declare associated EDG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ a 7 day supply of fuel.	In accordance with the Surveillance Frequency Control Program*

(continued)

* This Surveillance for ST-9AA and ST-9BB is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lube oil inventory of each EDG is ≥ a 7 day supply.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify Each EDG required air start receiver pressure is ≥ 150 psig.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program

* This Surveillance for ST-9AA and ST-9BB is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage on float charge is: a. ≥ 127.8 VDC for 125 VDC batteries, and b. ≥ 396.2 VDC for 419 VDC LPCI MOV independent power supply batteries.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.4.2	Verify each 125 VDC battery charger supplies ≥ 270 amps at ≥ 128 VDC for ≥ 4 hours. <u>OR</u> Verify each 125 VDC battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	- - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - This Surveillance shall not normally be performed in MODE 1, 2, or 3 for the 125 VDC batteries. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. - Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test or a modified performance discharge test.	In accordance with the Surveillance Frequency Control Program

(continued)

* This Surveillance for MST-071.10 is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

ACTIONS

CONDITIONS	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate action to restore required DC electrical power subsystem to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-----------------------------NOTE---------------------------- The following SRs are not required to be performed: SR 3.8.4.2, SR 3.8.4.3, and SR 3.8.4.4. -- For DC electrical power subsystem required to be OPERABLE the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, and SR 3.8.4.4.	In accordance with applicable SRs*

* This Surveillance for MST-071.12 is not required to be performed following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more batteries with average electrolyte temperature of the representative cells not within limits. <u>OR</u> One or more batteries with one or more battery cell parameters not within Category C limits.	B.1	Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	In accordance with the Surveillance Frequency Control Program*

(continued)

* This Surveillance for MST-071.10 and MST-071.12 is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	In accordance with the Surveillance Frequency Control Program*
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 65°F for each 125 VDC battery, and ≥ 50°F for each 419 VDC LPCI MOV independent power supply battery.	In accordance with the Surveillance Frequency Control Program*

* This Surveillance for MST-071.11 and MST-071.13 is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 342

EXELON FITZPATRICK, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

1.0 INTRODUCTION

By letter dated June 12, 2021, as supplemented by two letters dated June 13, 2021, and a letter dated June 14, 2021, that superseded all of the previous letters (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML21163A014, ML21164A000, ML21164A009, and ML21165A017, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted an emergency request for changes to the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) Technical Specifications (TSs). The proposed changes would modify technical specification (TS) 3.5.1, "ECCS [Emergency Core Cooling System] – Operating," TS 3.6.1.9, "Residual Heat Removal (RHR) Containment Spray System," and TS 3.6.4.1, "Secondary Containment", as well as certain Surveillance Requirements to support emergent repair of the "A" residual heat removal pump motor. Specifically, the amendment revises the Completion Time from 7 days to 34 days for the "A" residual heat removal pump, the Completion Time from 4 hours to 30 hours for restoring secondary containment, and the Completion Time from 7 days to 27 days for restoring one containment spray subsystem to operable status.[1] Additionally, the amendment allows extending the completion of several surveillance requirements of equipment being protected during the replacement of the "A" residual heat removal pump motor.

[1] The NRC staff acknowledges that there is a typographical error in the June 14, 2021 submittal regarding the one-time requested change in Completion Times for TS 3.5.1 and TS 3.6.1.9. See, e.g., Section 1.0, "Summary Description" of the LAR stating the change in Completion Time for TS 3.5.1 is "from 7 days to 30 34 days" and the change in Completion Time for TS 3.6.1.9 is "from 7 days to 23 27 days." The staff understands that the requested changes are accurately reflected in Attachment 2 of the LAR, "Markup of Technical Specification Pages" and in Section 2.8, "Description of the Proposed Change" of the June 14, 2021 submittal.

2.0 REGULATORY EVALUATION

2.1 System Description

Two redundant ECCS systems exist. The major equipment of the RHR system consists of two heat exchangers and four RHR pumps. One loop consisting of one heat exchanger, two RHR pumps in parallel, and associated piping is physically separated from the second loop to minimize the possibility of a single physical event causing the loss of the entire system. Provisions exist to cross connect the two loops of the RHR System by a single header making it possible to supply either loop from the pumps in the other loop.

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High-Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low-pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tanks (CSTs), they are capable of providing a source of water for the HPCI and CS systems.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems, each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the Reactor Pressure Vessel (RPV) via the corresponding recirculation loop. The two LPCI subsystems can be interconnected via the RHR System cross tie line; however, this line is maintained closed to prevent loss of both LPCI subsystems during a LOCA.

The Containment Spray functions, as described in the Technical Specifications (TS) Bases are to mitigate the effects of bypass leakage and to prevent the drywell temperature from exceeding its design value of 309 degrees Fahrenheit for a significant period of time in order to ensure the safety equipment can perform its associated function during a design basis event. Each subsystem consists of a suction line from the suppression pool, two RHR pumps, a heat exchanger, and its associated spray header embedded in and protected by the primary shield wall located in the drywell and to a common spray header suspended in the suppression chamber above the minimum water level.

2.2 Licensee Proposed TS Changes

The proposed amendment would provide the licensee 34 days to repair the “A” RHR pump motor. This would be accomplished by extending the Completion Times (CTs) for TS 3.5.1, 3.6.1.9 and 3.6.4.1, and changing the specified frequency of certain SRs on protected equipment.

The current CTs for Condition A of TS 3.5.1, 3.6.1.9 and 3.6.4.1 are 7 days, 7 days, and 4 hours, respectively. The licensee proposed a one-time amendment to add asterisks and corresponding footnotes to the CTs for Condition A of TS 3.5.1, 3.6.1.9 and 3.6.4.1. The footnote for the TS 3.5.1 Condition A CT would state:

> *The Completion Time to return the “A” RHR pump to OPERABLE is extended to 34 days, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only

change ending upon restoration of the “A” RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours.

The footnote for the TS 3.6.1.9 Condition A CT would state:

> *The Completion Time is extended to 27 days, in support of the “A” RHR pump repairs, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration of the “A” RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours.

The footnote for the TS 3.6.4.1 Condition A CT would state:

> *The Completion Time is extended to 30 hours, in support of the “A” RHR pump repairs, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration of the “A” RHR pump to OPERABLE, or on July 7, 2021 at 20:00 hours. Multiple entries may be necessary to implement compensatory actions, or to address unforeseen circumstances related to the “A” RHR pump motor replacement.

In Section 3.5 of the LAR the licensee also proposed asterisks and corresponding footnotes to the frequencies of SRs 3.3.3.2.1, 3.3.6.1.2, 3.3.6.1.4, 3.3.6.1.5, 3.5.1.1, 3.5.1.2, 3.5.3.1, 3.5.3.2, 3.5.3.3, 3.5.3.4, 3.6.1.3.5, 3.7.2.4, 3.7.4.1, 3.8.1.2, 3.8.1.3, 3.8.1.4, 3.8.1.5, 3.8.1.6, 3.8.3.1, 3.8.3.2, 3.8.3.4, 3.8.4.1, 3.8.5.1, 3.8.6.1, 3.8.6.2 and 3.8.6.3. These SRs are associated with equipment that would be protected during the motor replacement.

The footnote for SR 3.3.3.2.1 would state:

> *This Surveillance for ST-43I is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.3.6.1.2, 3.3.6.1.4, 3.3.6.1.5 would state:

> *This Surveillance for ISP-150B is not required to be performed for functions 4.a, 4.b, 4.d, 4.e and 4.f of table 3.3.6.1-1 until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.5.1.1 and 3.5.1.2 would state:

> *This Surveillance for ST-4B is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.5.3.1, 3.5.3.2, 3.5.3.3, 3.5.3.4 and 3.6.1.3.5 would state:

> *This Surveillance for ST-24J is not required to be performed until following the return of the “A” RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.7.2.4, 3.8.1.2 and 3.8.1.3 would state:

> *This Surveillances for ST-9BA and ST-9BB is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SR 3.7.4.1 would state:

> *This Surveillance ST-8Q is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.8.1.4, 3.8.1.5 and 3.8.1.6 would state:

> *This Surveillances for ST9AA, ST-9BB, ST-9BA and ST-9BA is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.8.3.1, 3.8.3.2 and 3.8.3.4 would state:

> *This Surveillance for ST-9AA and ST-9BB is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SR 3.8.4.1 would state:

> *This Surveillance for MST-071.10 is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SR 3.8.5.1 would state:

> *This Surveillance for MST-071.12 is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SR 3.8.6.1 would state:

> *This Surveillance for MST-071.10 and MST-071.12 is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

The footnote for SRs 3.8.6.2 and 3.8.6.3 would state:

> *This Surveillance for MST-071.11 and MST-071.13 is not required to be performed until following the return of the "A" RHR pump to OPERABLE. This past due Surveillance will be completed as stated in Section 3.5 of letter JAFP-21-0053, dated June 14, 2021.

2.3 Regulatory Requirements

Section 182a of the Atomic Energy Act (Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the license application. These TSs are derived from the plant safety analyses.

In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs.

The regulation at 10 CFR 50.36(b) requires:

> Each license authorizing operation of a …utilization facility …will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The regulation in 10 CFR 50.36(c)(2)(i) states, in part that:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. 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 regulation in 10 CFR 50.36(c)(3) states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The FitzPatrick TS 1.3, "Completion Times" establishes the CT convention and provides guidance for its use. Similarly, TS 1.4, "Frequency" defines the proper use and application of Frequency requirements. Usage rules for LCOs are in TS Section 3.0, "3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY." Per LCO 3.0.2, "Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met…"

Usage rules for SRs in TS Section 3.0, “SURVEILLANCE REQUIREMENT (SR) APPLICABILITY” dictate the requirements for SRs.

Per FitzPatrick SR 3.0.1:

> Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform the SR within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3.”

While the FitzPatrick TS Sections 1.0 and 3.0 are not regulations, they constitute license requirements imposed on plant operation.

Licensees may propose revisions to the TSs. The NRC staff reviews proposed changes and will generally issue changes provided that the plant-specific review supports a finding of continued adequate protection of public health and safety because: (1) the change is editorial, administrative, or provides clarification (i.e. no requirements are materially altered), (2) the change is more restrictive than the licensee’s current requirement, or (3) the change is less restrictive than the licensee’s current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, is discussed in Section 3.0 of this safety evaluation in the context of the proposed TS changes contained in the licensee’s LAR.

The regulations in 10 CFR 50.46(b) establish acceptance criteria for ECCS evaluations for light-water nuclear power reactors, as summarized below:

- Peak cladding temperature - the calculated maximum fuel element cladding temperature shall not exceed 2,200 degrees Fahrenheit (°F).

- Maximum cladding oxidation - the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

- Maximum hydrogen generation - the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Coolable geometry - calculated changes in core geometry shall be such that the core remains amenable to cooling.

- Long-term cooling - after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The applicable 10 CFR Part 50, Appendix A, “General Design Criteria [Criterion] for Nuclear Power Plants,” was considered as follows:

- Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

2.4 Regulatory Guidance

The NRC staff's guidance for the review of TSs is in Chapter 16.0, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [Light-Water Reactor] Edition" (SRP), March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications (STS) for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with the applicable reference STS (i.e., the current STS), as modified by NRC-approved travelers. The NRC used U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric BWR/4 Plants," NUREG-1433, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193, respectively).

The NRC staff also used guidance from the NUREG-0800 Section 6.3, "Emergency Core Cooling System," and Section 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary" for the review of this LAR.

3.0 TECHNICAL EVALUATION

3.1 Operation with One RHR Pump, One RHR Containment Spray Subsystem and Secondary Containment Inoperable

The proposed one-time extension of TS Completion Times includes TS 3.5.1 ECCS - Operating, Condition A, from 7 days to 34 days, TS 3.6.1.9 RHR Containment Spray System, Condition A, from 7 days to 27 days, and TS 3.6.4.1 Secondary Containment, Condition A, from 4 hours to 30 hours.

3.1.1 One RHR Pump and One RHR Containment Spray Subsystem Inoperable

The TS Bases state that if any one low pressure ECCS injection/spray subsystem is inoperable or if one LPCI pump in both LPCI subsystems is inoperable, the inoperable subsystem(s) must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single active component failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function.

Section 2.5 of the application stated that the Containment Spray system will remain available to perform its design function. As a result of not having a separate instrumentation specification in the TS, the system will be declared INOPERABLE because of the inability to measure flow to the Torus Spray header as described in the TS Bases for LCO 3.6.1.9. Section 3.1 of the application further states that the current status of the FitzPatrick RHR subsystem is that three RHR pumps remain available and capable of injecting to the vessel if a LOCA were to occur. In addition, both Core Spray pumps are OPERABLE, as well as HPCI and ADS. As discussed in Section 3.4 of the application, compensatory measures will be taken during the duration of the one-time extension of the TS Completion Time to gain additional margin for the ECCS and power system availability.

Section 16.3.2.2.5, "Emergency Core Cooling System," of the FitzPatrick Updated Final Safety Analysis Report (UFSAR) states that the most severe accident is that which is initiated by a ruptured main steam line. In turn, this event is assumed to cause failure of an adjacent Core Spray System and reactor recirculation riser line. Loss of the other core spray and two RHR pumps (electrical bus failure) is also assumed as the most severe single active failure acting concurrent with the postulated pipe failure. This then results in having only one LPCI loop utilizing two RHR pumps available to limit the core temperature rise. For the above conditions, analysis has shown that 10 CFR 50.46(b) acceptance criteria are satisfied including the peak clad temperature (PCT), which will be limited to 1370 degrees Fahrenheit.

The NRC staff reviewed the impact of the "A" RHR pump being out of service for the proposed extended period of time. Based on the deterministic evaluation discussed above and the compensatory measures the licensee is committed to take during this period, the NRC staff concludes that the plant will continue to meet the requirements of (1) 10 CFR 50.46(b), insofar as it requires that the acceptance criteria for the ECCS evaluations are met, and (2) GDC 35, insofar as it requires that a system to provide abundant emergency core cooling shall be provided, and that the system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. The NRC staff, therefore, finds the proposed emergency license amendment request acceptable.

3.1.2 Secondary Containment Inoperable

FitzPatrick contains four distinct radioactive material barriers, which includes systems, structures, and equipment, that together physically prevent the uncontrolled release of radioactive materials. The four barriers are the; (1) Reactor Fuel Barrier; (2) Reactor Coolant Pressure Boundary; (3) Primary Containment; and (4) Secondary Containment.

The LAR proposes to change the technical specification pertaining to the fourth barrier. In particular, the completion time of TS 3.6.4.1 will be increased from 4 hours to 30 hours for Condition A "Secondary containment inoperable in MODE 1, 2, or 3", which has a required action of "Restore secondary containment to OPERABLE status." This completion time of 30 hours is proposed to facilitate the emergent need for replacement of the "A" RHR pump motor. This motor replacement will require that the Reactor Building (RB) Crescent Area hatch be removed for a period of time up to 30 hours during Mode 1 of operation.

In particular, the proposed change to TS 3.6.1.4 will include a note that reads:

> **The Completion Time is extended to 30 hours, in support of the "A" RHR pump repairs, contingent on implementation of Compensatory Actions stated in Section 3.4 of letter JAFP-21-0053, dated June 14, 2021, as a one-time only change ending upon restoration of the "A" RHR pump to OPERABLE, or on July 11, 2021 at 20:00 hours. Multiple entries may be necessary to implement compensatory actions, or to address unforeseen circumstances related to the "A" RHR pump motor replacement.*

Exelon stated that the secondary containment functions as described in the Technical Specifications are:

1. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA).

2. Designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment in conjunction with operation of the Standby Gas Treatment (SGT) system and certain valves whose lines penetrate secondary containment.

3. Surrounds the primary containment and is designed to provide secondary containment for postulated loss-of-coolant accidents inside the primary containment.

4. Surrounds the primary containment and is designed to provide primary containment for the postulated refueling accident.

Due to the history of when FitzPatrick was constructed and commissioned, FitzPatrick is not bound by 10 CFR 50 Appendix A – General Design Criteria for Nuclear Plants, but instead has Principle Design Criteria.

FitzPatrick FSAR Section 1.5 "Principal Design Criteria" lists the following criteria as being relevant to the secondary containment and ECCS equipment contained therein:

> 1.5.6 Nuclear Safety Systems and Engineered Safeguard Criteria
>
> 1.5.6.1 General
>
> 5. Features of the plant which are essential to the mitigation of accident consequences are designed so that they can be fabricated and erected to quality standards which reflect the importance of the safety function performed.
>
> 6. The design engineered safeguards include allowance for environmental phenomena at the site.
>
> 1.5.6.2 Containment and Isolation Criteria
>
> 4. A secondary containment is provided that completely encloses both the primary containment and fuel storage areas.

5. The secondary containment is designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive barrier.
7. The primary and secondary containments, in conjunction with other engineered safeguards, act to prevent the radiological effects of accidents resulting in the release of radioactive material from the containment volumes from exceeding the guidelines of 10 CFR 100.

The NRC staff evaluated the issues of the Reactor Building hatch removal and its temporary impact on secondary containment; crescent area room cooling; and crescent area internal flooding. A discussion of the NRC staff's evaluation follows.

<u>Secondary Containment</u>

The NRC staff evaluated the potential impact of the open RB hatch on the functionality of Secondary Containment during the period(s) of time the RB Crescent Area hatch is removed.

Exelon stated:

> Repair of the A RHR pump requires a secondary containment breach for a period of time to support transit of equipment. The secondary containment is not designed to retain severe accident source terms and is reliant on the Standby Gas Treatment System (SGTS) to maintain a negative pressure in the Reactor Building. The SGTS system is designed to control release rates for the fuel handling accident and contain the release from a significant nuclear accident. In addition, panels installed on the Reactor Building are likely to detach and relieve pressure (0.5 psid) during severe accidents leading to a ground level release. Finally, hydrogen burns/deflagration are a significant challenge to reactor building isolation during severe accidents. Scrubbing of releases by Fire Protection Spray and/or steam released from primary containment could potentially provide a benefit in severe accidents, but such phenomena have little impact on noble gases and haven't been proven effective for other fission products. No irradiated fuel moves, nor any Dry-Cask evolutions will be performed during the requested extension.
>
> With these factors under consideration, the accident analysis essentially takes no credit for Reactor Building integrity when determining release rates. The impact of a short duration secondary containment breach has a negligible impact on releases to the public.

UFSAR Section 5.3.4 "Safety Evaluation" pertaining to Secondary Containment reads:

> The Reactor Building will be maintained at a negative pressure with respect to atmosphere by controlling the quantity of outside air introduced into the building with the two operating supply air fans in response to a signal from a differential pressure controller during normal operation. The two reactor building exhaust systems will operate at fixed air quantities.

UFSAR Section 9.3.3 "Reactor Building Ventilation System" reads in part:

> All exhaust air from the Reactor Building secondary containment is discharged to a common duct to the roof stack, which is equipped with dual radiation monitors

> and isolation butterfly valves (Section 7.12). Upon detection of high radiation by any one of the radiation monitors, the refuel floor exhaust fans are shutdown, the isolation butterfly valves will close, the tank exhaust fan will shutdown, and the supply and exhaust fans below the refuel floor will operate in a 100 percent recirculation mode. This condition is annunciated in the Control Room.

Based on UFSAR Section 5.3.4, with the RB West Crescent area hatch plug removed and the Reactor Building Ventilation system in normal operation there would be outside air influent into the area during normal power operations. Therefore, the Reactor Building Ventilation system exhaust system high radiation monitoring instrumentation controls the RB ventilation effluent during the period of time that the hatch is removed and provides monitoring of the building's release to the environment.

In the event that Reactor Building Ventilation system malfunctions, the compensatory measures associated with the proposed changes to TS 3.6.1.4 reads in part:

> Chemistry to establish a Low Volume sampler at the access hatch for continuous sampling and ensure sample results are collected every 12 hrs and analyzed within 24 hours of collection.

Therefore, the licensee has established a method of ensuring that no unmonitored releases to the environment will occur during the short duration of time that the hatch is removed during operation.

In addition, FitzPatrick will also be bound by the following compensatory measures during the time the West Crescent area hatch is removed:

- No fuel cask movement OR irradiated fuel handling operations are being performed in the Secondary Containment.

- No handling of loads over irradiated fuel.

- Section 5.3.2 "Safety Design Bases" pertaining to Secondary Containment reads in part:

  6. The Reactor Building is designed to be sufficiently leaktight to allow the Standby Gas Treatment System to reduce the reactor building pressure to a minimum subatmospheric pressure of 0.25 in. of water (under neutral wind conditions) when the Standby Gas Treatment System fans are exhausting Reactor Building atmosphere at a rate of 200 percent (3600 cfm) per day of the Reactor Building free volume.

FitzPatrick SR 3.6.4.1.4 reads "Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 6000 cfm."

It is noted that in the event a DBA would occur during the period of time the hatch is open, the Standby Gas Treatment System (SGTS) would initiate as designed, however, it is unknown whether the SGTS could reduce the reactor building pressure to a minimum subatmospheric pressure of 0.25 in. of water per SR 3.6.4.1.4 prior to restoration of Secondary Containment boundaries. However, with the SGTS in operation, the SGTS effluent would be processed by the SGTS safety related filter trains and the effluent discharge to the outside environment would be monitored for the level of radioactivity. Therefore, the SGTS will provide some level of mitigation.

In addition, with respect to the unlikely occurrence of a DBA happening with the hatch open, the compensatory measures established for TS 3.5.1 include provisions to "Perform shiftly briefs on the performance of local containment venting."

FitzPatrick will also be bound by the following compensatory measure during the time the West Crescent area hatch is removed:

> …Operations will be briefed on actions and response in the event of a loss of offsite power during the duration of the extension.
>
> - *"In the event of severe weather, including tornado/high wind events, all work associated with the equipment impacted by the changes proposed in the LAR will be suspended and the WC-AA-101, Attachment 7 for High Risk Evolutions for severe weather will be used to control work processes. Focus will also be placed upon weather forecasts as they pertain to the motor repair/replacement work to ensure that the removed floor plug opening is sealed prior to any potential rain."*

The NRC staff notes that the Loss of Offsite Power (LOOP) is a concern with respect to the loss of power to the "Spent Fuel Pool Cooling System" as this system is powered from a non-safety bus.

Given that the compensatory measures associated with the change to TS 3.6.1.4 provide the necessary direction to close the hatch before the potential LOOP occurs and before severe or hostile weather reaches FitzPatrick, the NRC staff finds the potential impacts to the FitzPatrick's Secondary Containment to be adequately managed to minimize the risk.

<u>Crescent Area Room Cooling</u>

The NRC staff evaluated the potential impact of the open RB hatch on the Crescent Area room cooling capabilities.

Exelon stated:

> … With regard to crescent area ventilation, TRM 3.7.C bases states that the function of the system is to maintain area temperatures below 110 degrees F [*sic* Fahrenheit*]* using lake water as the cooling medium up to a lake temperature of 85 degrees F. Upon reviewing the historical meteorological data for the Oswego, NY area, the highest recorded temperature in the month of July is 95 degrees F. Current lake temperature is 58.6 degrees F, with an outside air temperature of 66 degrees F. Considering this data, sufficient margin exists regarding safety margins for installed ventilations systems.

UFSAR Section 7.5.12 reads in part:

> The Crescent Area Coolers provide cooling to the Reactor Building El. 227 ft- 6in (Crescent Area). Analysis has shown, that with all five Crescent Area Coolers on a side in operation, the design temperature can be maintained, with a lake water

> injection temperature of 85°F. When use of all five crescent coolers in the same half of the crescent (rather than four) is required to remove the heat loads associated with elevated lake temperatures, single failure criteria is still met in that each half of the Crescent is 100% redundant to the other half in their ability to safely shut down the plant following a DBA event.

Moreover, the UFSAR Section 9.3.3.3 “Reactor Building Ventilation System” reads in part:

> At El 227 ft - 6 in, where safeguard equipment designed to operate during emergencies is installed, the outside supply air is sufficient to cool the area during normal operation. In an emergency, when the ventilation system is not functioning, the unit coolers, having sufficient capacity and quantity to satisfy the requirements of the emergency equipment installed at this level, are able to handle the entire heat gain load without outside air supply. The water supply to the cooling coils is from the Normal Service Water or Emergency Service Water System. Electric power is from the emergency buses. Unit coolers are installed with 100 percent redundancy. Redundant equipment is separately located in each half of this level and served by separate emergency power and water circuits.

Exelon indicated that FitzPatrick has three ongoing Adverse Condition Monitoring Plans (ACMPs) in place for the: HPCI Turbine Steam Inlet Isolation Valve; EHC Intercept Valve; and 6A Feedwater Heater Instrumentation. Crescent Area Room Cooling in not included within the current listing of ACMPs.

The FitzPatrick Technical Requirements Manual pertaining to the Crescent Area Ventilation System contains surveillance requirement TRS 3.7.C.2 which reads “Perform a comprehensive thermal performance test to quantify the maximum allowed lake inlet temperature such that five unit coolers can remove the designated heat loads with 110°F air temperature.” The minimum frequency of this surveillance is every 12 months.

The NRC staff notes that the compensatory measures established for the time limited use of TS 3.6.1.4 by the licensee includes “*Communications established, as required, with MMD* [Mechanical Maintenance Department] *to ensure a prompt restoration of the hatch is performed if requested by Operations*” in the advent of severe weather or in the event of a LOOP. Therefore, any deleterious effects of an open hatch on the Crescent Area room temperature would be remedied with the established compensatory measures.

Given the compensatory measures associated with the change to TS 3.6.1.4, provides the necessary direction to close the hatch when severe or hostile weather is eminent the NRC staff finds the potential impact to the FitzPatrick’s crescent area room cooling to be insignificant.

<u>Crescent Area Internal Flooding</u>

The NRC staff evaluated repair of the A RHR pump requires removal of the floor hatch above the West Crescent room. The NRC staff notes that the FitzPatrick East and West Crescent areas are a single cojoined area as displayed FSAR Figure No. 16.3-21.

Exelon stated:

> When assessing the impact to the plant, the relevant event of concern is internal flooding flood propagation from RB-272' to the West Crescent. RHR A, RHR C, RCIC, and Core Spray A are the key systems in the West Crescent. In terms of internal flooding potential, a stair tower with a normally closed fire door on RB-272' impedes flow between RB-272' and the West Crescent. This stair tower and un-isolated breaks on RB-272' can eventually impact equipment in the West Crescent.
> The open hatch additionally provides an expanded flow area that could reduce the time between Reactor Building 272' and above pipe ruptures and accumulation of water in the West Crescent.

UFSAR Section 6.5 "Safety Evaluation" pertaining to the: HPCI System; Automatic Depressurization System; Core Spray System and LPCI reads in part:

> Redundant portions of the Core Spray and LPCI Systems are protected from postulated flooding by a metal bulkhead located in the crescent area of the Reactor Building. Two 30 in. diameter spillways are provided to allow excess water to flow into the torus area. The spillways are sized to handle one LPCI pump discharge break, the limiting condition. The spillways also prevent reverse flow from the torus area to the crescent area, maintaining a watertight seal to one foot above the torus area water level.

The NRC staff notes that the application provides compensatory measures (i.e. "*flood protected*") to protecting hatch area while open, thereby mitigating the risk of flooding.

Exelon stated in the compensatory measures for TS 3.6.4.1 that communications would be established, with MMD to ensure a prompt restoration of the hatch is performed if requested by Operations.

- "Operator rounds will be established to ensure there are no leaks present within or waterflow that could propagate to the crescents that could jeopardize the operability of housed equipment. Spill barriers will surround the perimeter of the opened floor plug to further address water seepage into the west crescent."

- "In the event of severe weather, including tornado/high wind events, all work associated with the equipment impacted by the changes proposed in the LAR will be suspended and the WC-AA-101, Attachment 7 for High Risk Evolutions for severe weather will be used to control work processes. Focus will also be placed upon weather forecasts as they pertain to the motor repair /replacement work to ensure that the removed floor plug opening is sealed prior to any potential rain."

Exelon stated with regards to internal flooding, the rigging plan for the hatch removal and reinstallation meets NUREG-0612 requirements for double redundant rigging, mitigating the risk of a flooding incident imposed by the motor repair/replacement evolution. Furthermore, due to the ongoing rigging evolutions, personnel will be in the hatch area making manipulations and repairs while the hatch is open.

Given that the compensatory measures associated with the change to TS 3.6.1.4 provide the necessary direction to close the hatch when severe weather is eminent and that the potential of internal flooding is small during the short duration that the hatch would be removed, the NRC staff finds the potential impact to the FitzPatrick's internal flooding controls to be insignificant.

3.2 Risk Insights

While this is not a risk-informed LAR, the licensee discussed its risk assessment related to the proposed changes in Section 3.3 of the LAR. Because this is not a risk-informed LAR, the probabilistic risk assessment (PRA) models used by the licensee to derive risk insights were not reviewed by the NRC staff to determine their technical acceptability to support this safety evaluation.

However, the NRC staff considered the licensee-provided risk information in Section 3.3 of the LAR and associated risk management actions (RMAs) in Section 3.4 of the LAR to aid the deterministic review of the proposed change. The NRC staff did not rely on the quantitative risk metrics provided by the licensee in Section 3.3 of the LAR. The NRC staff determined that "special circumstances," as discussed in NUREG-0800, Standard Review Plan for Review of Safety analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (ADAMS Accession No. ML071700658), which would have necessitated additional risk information to be provided, did not exist for the proposed changes.

In addition to reviewing the licensee-provided information, the NRC staff reviewed the NRC's Standardized Plant Analysis Risk (SPAR) model for FitzPatrick to identify the dominant risk contributors and evaluate the risk insights for the proposed change. This review was performed considering the increased potential for common-cause failure of the RHR pumps. This review increases the NRC staff's confidence in the appropriateness of the licensee-provided compensatory measures. The NRC staff's review of the FitzPatrick SPAR model did not identify the need for any additional compensatory measures. The NRC staff's review also noted that the risk due to external hazards does not significantly change the risk from the proposed change because the licensee's compensatory measures include consideration of severe weather as well as external flooding and any additional impacts from external hazards such as seismic events are independent of the CT. The NRC staff's review supported the traditional engineering conclusions associated with the proposed change and the conclusion that the licensee's proposed RMAs support the availability of the remaining equipment to maintain defense-in-depth related to system redundancy, independence, and diversity.

The NRC staff's review also determined that the proposed one-time extensions to the completion of surveillance requirements listed in Section 3.5 of the LAR support the compensatory measures by (1) ensuring that mitigation equipment is available during the duration of the proposed change, and (2) reducing the risk from human error or maintenance error during the surveillance.

Based on the information provided by the licensee and the evaluation above, the NRC staff concludes that the available risk insights are acceptable for the purposes of supporting the deterministic evaluation.

3.3 Compensatory Measures

In Section 3.4 of its LAR, the licensee proposed Compensatory Measures that will be in place during the duration of the extended CTs. The licensee stated that it will place the measures to gain additional margin for ECCS and power system availability.

The NRC staff has reviewed the licensee's proposed compensatory measures and concludes, based on information provided by the licensee, that they are appropriate to reduce the risk of unnecessary plant transients, protect systems needed for accident mitigation, and raise operator awareness of necessary recovery actions with RHR, Core Spray and Secondary Containment systems inoperable to facilitate replacement of the "A" RHR pump motor.

3.4 Technical Specification Changes

The NRC staff reviewed the proposed TS changes to the affected CTs. The licensee provided an evaluation which stated:

> The ECCS system has sufficient capacity to function for design basis events while in Condition A. Assuming no additional failures, the UFSAR acceptance criteria for the design events will be met should such an event occur during the time that the "A" RHR pump is out of service.

In Section 3.4 of the LAR the licensee provided a list of Compensatory Measures that will be in effect for the duration of the one-time extension to the respective CTs.

The NRC staff reviewed the licensee's evaluation and found it acceptable because the extended CTs and footnotes are based on the justifications provided by the licensee in the application. That is, the extended CTs will only be available when the Compensatory Measures are in place and will only be available to support "A" RHR pump motor replacement activities and will not be available after 20:00 on July 11, 2021. Therefore, the NRC staff determined the regulatory requirements of 10 CFR 50.36 will continue to be met, because the TS will continue to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, in accordance with 10 CFR 50.36(b) and the TS will continue to require the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the LCO can be met, in accordance with 10 CFR 50.36(c)(2).

The NRC staff reviewed the proposed TS changes to the affected SRs. The licensee provided an evaluation and justification of the proposed footnotes for the affected SRs which stated:

> The basis for this request is to address the TS requirements. JAF [FitzPatrick] TS SR 3.0.1, states inpart "Failure to perform the SR within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3."
>
> Additionally, in order to maintain safe and stable plant operations as a result of the extended operation with the "A" RHR pump inoperable, JAF [FitzPatrick] is protecting other safety related equipment. SRs in protected systems or important to safety systems will come due, during "A" RHR motor replacement extended Completion Time, and if not performed the associated system must be declare[sic] inoperable.

The licensee further stated that it is requesting the footnotes to the selected SRs:

> … to minimize the likelihood of a human error or unforeseen circumstance arising from the operation of equipment that degrades other ECCS equipment or their associated power supplies.

> Review of each surveillance and the work orders open against the equipment associated indicate favorable history. System performance on each surveillance has shown to be reliable with repeatable results.

The NRC staff noted that given the requirements of the usage rules for LCOs and SRs described in Section 2.0 above, the licensee must perform SRs within the respective specified frequencies. However, for the equipment listed as protected in the Compensatory Measures, performance of SRs may be inappropriate during the time the Compensatory Measures are needed. The NRC staff reviewed the FitzPatrick TS Section 1.4 "Frequency" as well as the corresponding section in NUREG-1433. The FitzPatrick TS Section 1.4 is essentially identical to the corresponding section in NUREG-1433 and both contain a discussion of situations where a SR could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed.

The NRC staff determined the footnotes modify the performance requirements of the SRs and are therefore part of the specified frequency of the affect SRs. This type of situation is described in TS Section 1.4. The footnotes create a one-time extension of the interval specified in the existing respective Frequency columns. The NRC staff determined these extensions are warranted given the plant operating conditions are not suitable for conducting the SRs. The NRC staff further determined the extensions do not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency because of the recognition that the most probable result of any particular SR being performed is the verification of conformance with the SRs. The one-time nature of the extensions is assured by the reference in the footnote to Section 3.5 of the licensee's application. That section contains the rescheduled dates for the performance of the respective SRs.

Even though the footnotes create a one-time extension of the interval specified in the respective Frequency columns for the selected SRs, the extension conforms with SR usage rules and all other SR and LCO usage rules remain unchanged. Per SR 3.0.1, failure to meet a SR, whether such failure is experienced during the performance of the SR <u>or between performances of the Surveillance</u> (emphasis added), shall be failure to meet the LCO. So, if information becomes available which leads to a conclusion a particular SR is not met, the associated LCO would be required to be declared not met.

Therefore, NRC staff also determined the SRs, as amended by the proposed changes will continue to meet 10 CFR 50.36(c)(3) because the SRs will still provide assurance that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Finally, the NRC determined the proposed changes to the selected SRs can allow the licensee to remain in compliance with TS usage rules contained in the FitzPatrick TS during the "A" RHR pump motor replacement activities.

Consistent with NUREG-0800, special attention is given to TS provisions that depart from the reference TS (STS) where the differences from the STS can be justified by other considerations so that 10 CFR 50.36 is met. The NRC staff further notes that while similar footnotes do not exist in NUREG-1433, the proposed changes requested in this emergency LAR provides more appropriate CTs and specified SR frequencies commensurate with the risk associated with continued plant operation during "A" RHR pump motor replacement activities. Given the one-time nature of the condition, the NRC staff finds this departure from the STS is acceptable. Therefore, the NRC staff determined the proposed change to the TS is acceptable.

3.5 Region Staff Verification of SR Historical Performance

The inspectors performed a 24-month review of past performances associated with the 24 specific SRs listed in Section 3.5 of the application that would otherwise have to be performed on equipment proposed to be protected during the extended period. The inspectors also performed a review of the past 24 months to assess the performance of the equipment that would be protected during the extended period. The inspectors also assessed adverse condition monitoring plans to ensure operable, but degraded equipment remains available to support its safety function. The inspectors reviewed upcoming work schedules to ensure no maintenance would be performed on protected equipment. The inspectors reviewed compensatory measures associated with protection of equipment and fire mitigation; and performed walkdowns to ensure the equipment remained available. The inspectors did not identify any immediate concerns with the systems or components for which SRs may be extended. The inspectors also did not identify any immediate concerns in the corrective action program related to equipment performance of those systems that will be protected during the extended period.

After reviewing the information above, the NRC staff finds that postponement of the SRs during the extended CT as proposed by the licensee will maximize the availability of the protected equipment and support 10 CFR 50.36(c)(3) during the extended CT since the necessary quality of the systems and components is maintained.

3.6 Technical Conclusion

Based on the information provided by the licensee and the analysis in Section 3.0 of this safety evaluation, the NRC staff concludes that while the licensee's proposed TS changes are less restrictive than the licensee's current TS requirements, the proposed changes still provide adequate assurance of safety when judged against current regulatory standards. The licensee's proposed CT extensions do not have any impact on the licensee's compliance with the 10 CFR, Appendix A General Design Criteria (as described in the licensee's Updated Final Safety Analysis Report), 10 CFR 50.46 ECCS performance criteria, or the 10 CFR 50.36 TS requirements. The NRC staff also concludes that the available risk insights are acceptable for the purposes of supporting the deterministic evaluation for the proposed one-time changes. Therefore, the extension of the CTs and changes to SRs associated with the inoperability of the "A" RHR pump is acceptable.

4.0 EMERGENCY SITUATION

The NRC's regulations in 10 CFR 50.91(a)(5) state that where the NRC finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the NRC will publish a notice of issuance under 10 CFR 2.106, providing for opportunity for a hearing and for public comment after issuance.

As discussed in the licensee's application dated June 12, 2021, as supplemented by two letters dated June 13, 2021, and a letter dated June 14, 2021, the licensee requested that the proposed amendment be processed by the NRC on an emergency basis. The licensee stated, in part, in its application:

> [T]he emergency circumstances resulted from the unforeseen failure of the JAF [FitzPatrick] "A" RHR pump during its return to operability during the scheduled maintenance window. The required Completion Time for Condition A for TS 3.5.1 of seven days is currently applicable and will expire on June 14, 2021 at 20:00 EDT. JAF [FitzPatrick] cannot complete the required repair and TS required testing by that time, and neither a routine nor an exigent amendment can be processed prior to June 14, 2021 at 20:00 EDT.
> A review of maintenance history of the "A" RHR Pump/Motor was performed. The data collected as part of the prescribed preventative and predictive maintenance performed from 2014 until present showed no indication of degradation. The last offline motor test was performed was in 2014 with satisfactory results.
>
> JAF [FitzPatrick] requests an expedited review of the proposed license amendment in accordance with the provisions of 10 CFR 50.91(a)(5) based on avoiding the need to shut down the JAF [FitzPatrick] Unit without an approved amendment. If the proposed license amendment is not approved, JAF [FitzPatrick] will be required to enter TS 3.5.1 Condition B and commence a shutdown on June 14, 2021 at 20:00 EDT.
>
> JAF [FitzPatrick] has determined that emergency circumstances exist, has used its best efforts to make a timely application, and did not knowingly cause the emergent situation.

The NRC staff reviewed the licensee's basis for processing the proposed amendment as an emergency amendment (as discussed above) and determines that an emergency situation exists consistent with the provisions in 10 CFR 50.91(a)(5). Furthermore, the NRC staff determined that: (1) the licensee used its best efforts to make a timely application; (2) the licensee could not reasonably have avoided the situation; and (3) the licensee has not abused the provisions of 10 CFR 50.91(a)(5). Based on these findings, and the determination that the amendment involves no significant hazards consideration as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendment using the emergency provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), by letter dated June 12, as supplemented by two letters dated June 13, 2021, and a letter dated June 14, 2021, the licensee provide its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change involves a one-time extension to the Completion Time for Technical Specification (TS) 3.5.1 Condition A to allow necessary time to restore the "A"

RHR pump to OPERABLE status. The proposed amendment does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed amendment does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. The proposed amendment will not alter assumptions relative to the mitigation of an accident or transient event. Furthermore, the Emergency Core Cooling Systems (ECCS) will remain capable of adequately responding to a design basis event or transient during the period of the extended Completion Time.
Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different accident from any accident previously evaluated?

   Response: No

   The proposed amendment does not introduce any new or unanalyzed modes of operation. The proposed changes do not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis.

   Therefore, the proposed amendment will not create the possibility of a new or different accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers is not affected by the proposed amendment; therefore, the margins to the onsite and offsite radiological dose limits are not significantly reduced.

   In addition, during the extended Completion Time the ECCS will remain capable of mitigating the consequences of a design basis event such as a LOCA.

   Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on its review of the licensee's evaluation above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on June 12, 2021. The State official had no comments.

7.0 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 and changes surveillance requirements. 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. 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.

8.0 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.

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Date: June 14, 2021

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 342 RE: ONE TIME EXTENSION TO TECHNICAL SPECIFICATIONS 3.5.1, 3.6.1.9, AND 3.6.4.1 COMPLETION TIMES TO SUPPORT RESIDUAL HEAT REMOVAL PUMP MOTOR REPLACEMENT **(EMERGENCY CIRCUMSTANCES)**(EPID L-2021-LLA-0110) DATED JUNE 14, 2021

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