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

September 4, 2024

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT NO. 356 RE: UPDATE FUEL HANDLING ACCIDENT ANALYSIS (EPID L-2023-LLA-0109)

Dear David Rhoades:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 356 to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment revises the technical specifications in response to your application dated August 3, 2023, as supplemented by letters dated August 31, 2023, February 28, 2024, March 25, 2024, and July 29, 2024.

The amendment revises the technical specifications to change the fuel handling accident analyses in support of the transition from the refuel bridge mast NF-400 (*i.e.,* triangular mast) to the new NF-500 mast.

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

Sincerely,

/RA/

James S. Kim, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Amendment No. 356 to DPR-59
2. Safety Evaluation

cc: Listserv

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

CONSTELLATION FITZPATRICK, LLC

AND

CONSTELLATION ENERGY GENERATION, LLC

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 356
Renewed Facility Operating License No. DPR-59

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

   A. The application for amendment by Constellation Energy Generation Company, LLC, dated August 3, 2023, as supplemented by letters dated August 31, 2023, February 28, 2024, March 25, 2024, and July 29, 2024, 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) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 356, 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 its date of issuance and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

HIPOLITO GONZALEZ

Digitally signed by HIPOLITO GONZALEZ
Date: 2024.09.04 10:54:54 -04'00'

Hipólito González, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 4, 2024

ATTACHMENT TO LICENSE AMENDMENT NO. 356

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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

Remove Page	Insert Page
Page 3	Page 3

Replace the following page 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 Page	Insert Page
3.3.6.2-4	3.3.6.2-4
3.3.7.1-1	3.3.7.1-1
3.6.4.1-1	3.6.4.1-1
3.6.4.2-1	3.6.4.2-1
3.6.4.3-1	3.6.4.3-1
3.7.3-1	3.7.3-1
3.7.4-1	3.7.4-1
3.8.2-1	3.8.2-1
3.8.5-1	3.8.5-1
3.8.8-1	3.8.8-1

(3) Constellation Energy Generation, LLC, 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 as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Constellation Energy Generation, LLC, 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) Constellation Energy Generation, LLC, 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) <u>Maximum Power Level</u>

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

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 356, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Reactor Vessel Water Level – Low (Level 3)	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5 SR 3.3.6.2.6	≥ 177 inches
2.	Drywell Pressure - High	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5 SR 3.3.6.2.6	≤ 2.7 psig
3.	Reactor Building Exhaust Radiation - High	1, 2, 3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.6	≤ 24,800 cpm
4.	Refueling Floor Exhaust Radiation - High	1, 2, 3, (a)	1	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.6	≤ 24,800 cpm

(a) During movement of recently irradiated fuel assemblies in secondary containment. “Recently irradiated” is defined for Technical Specification 3.3.6.2 as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.

3.3 INSTRUMENTATION

3.3.7.1 Control Room Emergency Ventilation Air Supply (CREVAS) System Instrumentation

LCO 3.3.7.1 The Control Room Air Inlet Radiation – High channel shall be OPERABLE.

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

---------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.3.7.1 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Channel inoperable.	A.1	Place the CREVAS System in the isolate mode of operation.	1 hour
	<u>OR</u>		
	A.2	Declare both CREVAS subsystems inoperable.	1 hour

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.

------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.6.4.1 as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.

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.

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

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

-----------------------------------NOTE------------------------------
“Recently irradiated” is defined for Technical Specification 3.6.4.2 as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.

ACTIONS

---NOTES---
1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by inoperable SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. <u>AND</u>	8 hours (continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

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

---------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.6.4.3 as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in during movement of recently irradiated fuel assemblies in the secondary containment.	---------------NOTE-------------- LCO 3.0.3 is not applicable. ----------------------------------- C.1 Place OPERABLE SGT subsystem in operation. <u>OR</u>	Immediately (continued)

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Ventilation Air Supply (CREVAS) System

LCO 3.7.3 Two CREVAS subsystems shall be OPERABLE.

---------------------------------NOTE------------------------------
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

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

---------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.7.3 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVAS subsystem inoperable for reasons other than Condition B.	A.1 Restore CREVAS subsystem to OPERABLE status.	7 days
B. One or more CREVAS subsystems inoperable due to inoperable CRE boundary in MODE 1, 2, or 3.	B.1 Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	<u>AND</u>	
	B.3 Restore CRE boundary to OPERABLE status.	90 days

(continued)

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

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

---------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.7.4 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1 Restore control room AC subsystem to OPERABLE status.	30 days
B. Two control room AC subsystems inoperable.	B.1 Verify control room area temperature < 90°F. <u>AND</u> B.2 Restore one control room AC subsystem to OPERABLE status.	Once per 4 hours 72 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and one division of the plant Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, “Distribution Systems-Shutdown”;

b. One qualified circuit, which maybe the same circuit required by LCO 3.8.2.a, between the offsite transmission network and the other division of the plant Class 1E AC electrical power distribution subsystem(s), when a second division is required by LCO 3.8.8; and

c. One emergency diesel generator (EDG) subsystem capable of supplying one division of the plant Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

---------------------------------NOTE------------------------------
“Recently irradiated” is defined for Technical Specification 3.8.2 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 One 125 VDC electrical power subsystem shall be OPERABLE to support one division of the plant Class IE DC Electrical Power Distribution System required by LCO 3.8.8, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

-------------------------------NOTE-------------------------------
"Recently irradiated" is defined for Technical Specification 3.8.5 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

ACTIONS

-------------------------------NOTE-------------------------------
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required DC electrical power subsystem inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
		(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems - Shutdown

LCO 3.8.8 The necessary portions of the AC and 125 VDC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
During movement of recently irradiated fuel assemblies in the secondary containment.

---------------------------------NOTE------------------------------
"Recently irradiated" is defined for Technical Specification 3.8.8 as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or 125 VDC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
		(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 356

CONSTELLATION FITZPATRICK, LLC

CONSTELLATION ENERGY GENERATION, 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 August 3, 2023 (Agencywide Documents Access and Management System Accession No. ML23215A012) as supplemented by letters dated August 31, 2023, February 28, 2024, March 25, 2024, and July 29, 2024 (ML23243A946, ML24059A130, ML24085A233, and ML24211A122, respectively), the Constellation Energy Generation, LLC (Constellation, the licensee), submitted a license amendment request to modify James A. FitzPatrick Nuclear Power Plant (FitzPatrick) Technical Specifications (TSs) to change the fuel handling accident (FHA) analysis in support of the transition from the Refuel Bridge Mast NF-400 (i.e., Triangular Mast) to the new NF-500 Mast. These changes are in support of the planned utilization of the NF-500 mast for refueling starting with a refueling outage scheduled for September 2024.

On October 31, 2023, the Nuclear Regulatory Commission (NRC) staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (88 FR 74530) for the proposed amendment. Subsequently, by letters dated February 28, 2024, March 25, 2024, and July 29, 2024, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the Nuclear Regulatory Commission (NRC) published a second proposed NSHC determination in the *Federal Register* on August 2, 2024 (89 FR 63226), which superseded the original notice in its entirety.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulatory Requirements

Under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a) (regarding, among other things, consideration of the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals) and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public, and that the applicant will comply with the Commission's regulations.

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met.

The regulations in 10 CFR 50.67, “Accident source term,” require, in part, in § 50.67(b)(2) that the applicant’s analysis must demonstrate with reasonable assurance that: (i) an individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 0.25 Sv (25 rem) [roentgen equivalent man] Total Effective Dose Equivalent (TEDE); (ii) an individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE; and (iii) adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Appendix A to Part 50, “General Design Criteria for Nuclear Power Plants,” Criterion 19, “Control Room” (GDC 19), states, in part, that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. GDC 19 further states that holders of operating licenses using an alternative source term under § 50.67 shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) TEDE as defined in § 50.2 for the duration of the accident.

NUREG-0800, Chapter 18, “Human Factors Engineering [HFE],” Revision 3, issued December 2016 (ML16125A114), provides the NRC staff guidance on reviewing HFE engineering practices and guidelines.

NUREG-0800, Standard Review Plan (SRP) Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases” (ML052340583), covers review of atmospheric transport and diffusion models to calculate relative concentrations for postulated accidental radioactive and hazardous airborne releases meteorological data summaries used as input to diffusion models, and derivation of diffusion parameters.

Regulatory Guide (RG) 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants” (ML071080736), provides guidance concerning criteria for an onsite meteorological measurements program that the NRC staff considers acceptable for the collection of basic meteorological data needed to support plant licensing and operation.

RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors” (ML003716792), provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This regulatory guide provides more conservative TEDE dose criteria for certain design basis accidents including the fuel handling accident. As specified in Table 6 “Accident Dose Criteria” in RG 1.183 Rev. 0, the exclusion area boundary (EAB) and the low population zone (LPZ) dose criteria for a fuel handling accident is 6.3 rem TEDE.

RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants” (ML031530505), provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants.

2.2 Proposed Technical Specifications Changes

The following TS changes were submitted by the licensee in its supplement dated July 29, 2024 (ML24211A122):

2.2.1 The proposed change adds the following Note in the ‘Applicability’ statement of TS 3.6.4.1, Secondary Containment, TS 3.6.4.2, Secondary Containment Isolation Valves (SCIVs), and TS 3.6.4.3, Standby Gas Treatment (SGT) System:

> - NOTE- - - - - - - - - - - - - - - - - - - -
> “Recently irradiated” is defined [for TSs listed above], as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.
> -

2.2.2 The proposed change adds the following text (as shown in bold text) in footnote (a) of Table 3.3.6.2, Secondary Containment Isolation Instrumentation:

> During movement of recently irradiated fuel assemblies in secondary containment. “**Recently irradiated” is defined for Technical Specification 3.3.6.2 as fuel assemblies which have occupied part of a critical reactor core within the previous 24 hours.**

2.2.3 The proposed change adds the following Note in the ‘Applicability’ statement of TS 3.3.7.1, the Control Room Ventilation (CREVAS) System Instrumentation, TS 3.7.3, CREVAS System, TS 3.7.4, Control Room Air Conditioning (AC) System, and TS 3.8.2, Electrical Power Systems/AC Sources — Shutdown:

> - NOTE- - - - - - - - - - - - - - - - - - - -
> “Recently irradiated” is defined [for TSs listed above], as fuel assemblies which have occupied part of a critical reactor core within the previous 104 hours.
> -

3.0 TECHNICAL EVALUATION

3.1 Description of Personnel Actions

The proposed changes involve adding to certain LCOs a definition of "recently irradiated" fuel that varies from LCO to LCO. For each of the relevant LCOs, the definition is provided in a note. As explained below, usage of the LCO-specific definitions assure that the LCOs will reflect at least the lowest functional capability or performance levels of equipment required for safe operation, as analyzed in the FHA. The definitions added to the TS reflect how fuel handling analyses are considered in operability of the Secondary Containment (SC) and the CREVAS operability, along with revisions to the associated FHA. For SC operability, the definition of recently irradiated fuel will be changed from the current 96 hours down to 24 hours. For CREVAS system operability, the definition of recently irradiated fuel will be changed from the current 96 hours up to 104 hours.

The FHA analyses that support these proposed changes involve three cases described below.

Case 1 accounts for the scenario from 0 – 24 hours after reactor shutdown with SC operable. SC actuation occurs by 2-minute automatic actuation, consistent with the existing licensing basis. While CREVAS would be operable during Case 1, it is not credited in the analysis.

Case 2 accounts for the scenario from 24 - 104 hours after reactor shutdown where SC is not operable, but CREVAS is operable. CREVAS action occurs by manual operator action as controlled by existing operator procedure AOP-44 [Abnormal Operating Procedure] "Dropped Fuel Assembly." Crediting of CREVAS as early as 24 hours after reactor shutdown requires an 8-minute Time Critical Action after initiation of the event to isolate control room ventilation. The operator action is defined by the following criteria:

- Procedure is entered when the control room operator in communication with the refuel floor reports a fuel handling accident.
- Success criteria is achieved when the control room is isolated by manual switch in the control room and a supply isolation bypass damper is manually positioned closed to address single failure of inlet isolation valve.

This revision to the required time for the time critical operator action will be managed per guidance in OP-AA-102-106, "Operator Response Time Program."

Case 3 accounts for the scenario greater than 104 hours after reactor shutdown where SC and CREVAS are no longer credited and therefore are not required to be operable. No automatic or manual actions are modeled within Case 3.

The final dose results for all three cases and associated applicable dose criteria are shown in Table 2 for GNF2 fuel and Table 3 for GNF3 fuel. This analysis is based on the dose criteria of 5 rem TEDE in the Control Room for the duration of the event, 6.3 rem TEDE at the EAB for the worst 2 hours, and 6.3 rem, TEDE at the outer boundary of the LPZ for the duration of the event:

Table 2 (FitzPatrick FHA Results – Proposed Changes for GNF2 Fuel)

	30-day CR TEDE (rem)	Max 2-hour EAB TEDE (rem)	30-day LPZ TEDE (rem)
Case 1	2.13	2.98	0.333
Case 2	4.95	0.541	0.0605
Case 3	4.94	0.285	0.0319
Limit	5	6.3	6.3

Table 3 (FitzPatrick FHA Results - Proposed Changes for GNF3 Fuel)

	30-day CR TEDE (rem)	Max 2-hour EAB TEDE (rem)	30-day LPZ TEDE (rem)
Case 1	2.03	2.84	0.317
Case 2	4.71	0.515	0.0576
Case 3	4.71	0.271	0.0303
Limit	5	6.3	6.3

3.2 Task Analysis

In the letter dated February 28, 2024, the licensee provided a description of task analyses conducted for operator actions affected by the proposed change.

The operator manual action is initiated by both symptom-based and event-based criteria associated with a refueling accident. Technical Specification required instrumentation will remain operable for Control Room Ventilation instrumentation. Redundant to this, continuous communication between the refueling bridge and the control room is established per OSP-66.001, Management of Refueling Activities, during all fuel transfer operations. The fundamental initiating condition is based on the dropped fuel bundle event.

AOP-44, Dropped Fuel Assembly, directs the isolation of control room ventilation. These actions involve performing a single switch manipulation in the backpanels of the main control room, verifying that the appropriate isolation dampers have closed and that the emergency ventilation fans have started using the indication available at the control room panel as described by RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.” To prevent loss of function on a single failure of the supply isolation damper, a redundant isolation bypass damper is then manually positioned closed.

Access to this damper involves exiting the control room and traversing the same level of the Admin Building elevation, a Seismic Class II structure, to the Admin Building Fan room, up a twelve-foot ladder, and to the bypass damper. Success criteria is conservatively assumed to be achieved upon closure of this manual bypass damper. Performing these actions as directed by abnormal operation procedures is an evaluated task in non-licensed operator training. FitzPatrick Technical Specifications require one non-licensed operator during modes 4 and 5, when this time critical action would be required. Emergency and administrative staffing requirements would add significant margin in staffing during modes 4 and 5.

Control room actions would be expected to be directly supervised by the control room supervisor, presenting an opportunity for validation of successful performance by another operator. These control room actions and indications are adequate to achieve the design basis isolation, however single failure proof manual closure of the bypass damper would be performed by an operator outside the control room in the manner described above.

The licensee concluded based on this review and the reduction in available time that while the existing action to isolate the control room is time validated at 6.5 minutes, it is prudent to streamline the AOP guidance to increase margin. The licensee plans to update the AOP guidance prior to implementing the license amendment.

The licensee has described the actions needed to operate CREVAS that have been validated and a timing analysis has been conducted. These actions are directed by procedures and are included in the licensee's training program. In addition, the tasks will be directly supervised by a Control Room Supervisor to ensure that they are carried out per procedure. The NRC staff has determined that these are acceptable to ensure that the system can be operated safely within the allotted time.

3.3 Timing Analysis for Operator Actions

In the letter dated February 28, 2024, the licensee stated the following in its responses to NRC questions:

- Please provide a description of the results of the timing validation of the new time critical operator action for verification of CREVAS manual actuation.

  No new operator action has been created as a result of this amendment. The amendment revises the assumed time of completion for an existing licensing basis operator action to isolate the control room, which has previously been validated per OP-AA-102-106 at 6.5 minutes. While enhancements are planned to increase the margin to the required completion time, these actions are not required to bring the existing action into compliance with the 8 minute assumptions described in the License Amendment Request (LAR).

- Measures included to create realistic scenario conditions:

  The Validation Team Leader directs the plant walkthrough by first providing the plant initial condition and then providing appropriate cues while the personnel walk through each procedure step (e.g., that fan indication indicates red light on, green light off). The validation personnel use the procedures in accordance with the scenario and plant walkthrough or talk through actions they would take in response to each instruction step. During the plant walkthrough, personnel performing the procedures describe the actions they are taking, identify the information sources used to take actions, identify the controls used to carry out actions expected system responses, how responses were verified, and the actions to be taken if responses did not occur.

- Any issues identified with procedural completeness, technical accuracy, and usability:

  There were no issues identified with procedure completeness, technical accuracy, and usability.

- Any training program issues identified:

  There were no training program issues identified.

- Whether the credited operator actions could be completed within the allowed time and whether adequate margin exists between the time required and time allowed:

  The isolation of control room ventilation is currently time validated per OP-AA-102-106 as an established Operator Time Critical Action with a performance of 6.5 minutes. Based on the risk significance of the task, adequate margin currently exists to the proposed required completion time of 8 minutes. Based on the significant reduction in the allowable time, prudent actions to streamline the implementing procedure are in progress to increase this margin.

- Whether any complicating factors that might be expected to affect the crew's ability to perform the credited operator actions were included:

  No required complicating factors are included in the validation of this time critical action, however, for one of the periodic validations, an injected complicating factor of a damper failure to operate and the associated delays was encompassed, and the activity still successfully performed within 6.5 minutes.

- How many complete crews of operators participated in the walkthrough scenarios:

  For this amendment, the scope of the task required to meet the design basis assumptions was revisited and the time subsequently revalidated at between 4 and 5 minutes using two independent equipment operators per the guidance described above. With an allowable time (AT) of 8 minutes, the smallest amount of margin we were left with following isolation is 37.5 percent.

  As a result of the greater than 20 percent margin, the Operator Response Time Program per OP-AA-102-106 did not require validating the action a third time.

  There are no new operator actions as a result of this proposed change. The licensee conducted a comprehensive timing analysis and validation to verify that operators could take the intended actions within the newly developed times. During this validation process, no new training requirements or procedure changes were identified. Additionally, operators participated in walkthrough scenarios in order to verify that the actions could be taken within the allotted timeframe. The NRC staff has determined that the change to the timing for the time critical action is acceptable.

3.4 Training

The licensee stated in its February 28, 2024, response to the NRC staff's February 1, 2024, request for additional information (RAI) that consistent with the existing licensing basis, operators are required to manually isolate the control room upon a design basis refueling accident. The pre-existing time critical action to isolate the control room is an evaluated task in equipment operator training, screened in accordance with industry standard procedure NISP-TR-01, Systematic Approach to Training (SAT) Process, as a low risk significance, a high frequency of routine performance and a low difficulty. Control/Relay Room Ventilation System is trained to Equipment Operator Continuing Training every 3 years per the Long-Range Training

Plan and was last taught in 2022. Any changes to the implementing procedures made to establish additional margin will be assessed for impact on existing training. No change in operator training is required at this time based on this reduction in the available time of completion.

The NRC staff has determined that this is acceptable based on the licensee's use of the NRC approved SAT process. Any changes to procedures will be evaluated for potential impacts on training activities.

3.5 Staff Evaluation

The licensee describes that the human action is starting CREVAS within 8 minutes if an FHA occurs in order to limit dose to the control room operators. The action would occur during refueling when the reactor and fuel is cold and the cavity is full of water. If this action failed, there would be no risk impact because an FHA does not result in core damage (no core melt because the fuel is cold and still surrounded by water) and does not contribute to large early release frequency, which are the Probabilistic Risk Assessment measures for risk.

As discussed below in Section 3.6, the NRC staff concludes that the information provided in section 3.0 of the LAR, supported by the information provided in the February 28, 2024, RAI response, demonstrates how the LAR, if approved, will not adversely impact CREVAS's ability to function and perform as needed in providing assistance to maintain control room operator doses within acceptable limits.

The licensee also discussed the timing analysis of the time critical operator action as well as the administrative controls and procedures necessary to timely complete the credited actions to maintain control room ventilation. The proposed changes which affect TEDE as discussed above still remain below the limits in the regulations. In addition, there is no new operator action being proposed.

3.6 Fuel Handling Accident (FHA)

The licensee evaluated the consequences of this event. The FHA analysis postulates that a spent fuel assembly is dropped during refueling. This accident is postulated to occur inside the containment as FitzPatrick does not have a separate fuel building. For this analysis the licensee evaluated two different fuel types that are part of their current licensing basis.

The inventory of fission products in the reactor core is a function of the reactor power, the duration of the at-power operation, and the time after shutdown prior to spent fuel movement. FitzPatrick determined the core inventory assuming a power level of 2,587 megawatt thermal rated power (102 percent of the rated thermal power). To account for differences in power distribution across the core, a radial peaking factor of 1.7 is applied to the average inventory. The majority of the fission products produced during operation are contained within the fuel pellet, however, some migrate to void spaces, known as "gap", within the fuel rods.

This LAR assumes no decay time prior to fuel movement. This assumption is in line with the current licensing basis (CLB), but this assumption of no decay time was changed under the 10 CFR 50.59 process. The previous docketed FHA licensing basis that was approved by the NRC (ML022350228) assumed a minimum of 96 hours before fuel handling could commence. However, shortly after that, the licensing basis was updated by the 10 CFR 50.59 process to contain two cases based on total decay time. The first case assumed no decay time but with an isolated and operable secondary containment, while the second case assumed 96 hours of decay but inoperable secondary containment.

This FHA analyzes three different scenarios/time frames depending upon when the fuel handling accident occurs. The first scenario postulates a fuel handling accident that occurs immediately after reactor shutdown, where fuel decay time equals 0 hours extending to 24 hours post-shutdown. The second scenario analyzes the dose consequences during a fuel handling accident that occurs 24 hours post-shutdown, to 104 hours post-shutdown. The third scenario analyzes the dose consequences 104 hours post-shutdown and greater. In order for defueling to occur from hour 0 after shutdown and meet dose acceptance criteria, the SC operability must be maintained which allows for crediting of the Standby Gas Treatment System (SGTS) after a 2-minute delay for actuation. For the time period between 24 hours and 104 hours, SC operability is not credited nor is the associated SGTS (to include both SGTS subsystems) and therefore they do not need to be operable. However, the CREVAS system after manual operator action is credited and must be operable during this time period. From hour 104 until the end of the accident analysis, neither SC (and SGTS) nor CREVAS are credited, and therefore these systems do not need to remain operable.

The mechanical part of FitzPatrick's analysis changes from the CLB. This new analysis supports the use of GNF2 and GNF3 fuel types. The source terms were determined using the ORIGEN-ARP code as noted in RG 1.183 (Revision 0) Section 3.1 "Fission Product Inventory." A fuel handling accident over the reactor vessel and an accident when the fuel bundle reaches the spent fuel pool were analyzed and compared. The accident over the reactor vessel would result in a greater release of the available source term due to a significantly higher number of damaged fuel rods and is the bounding event.

For the purposes of establishing the quantity of fuel damaged in the event, FitzPatrick assumes that a Global Nuclear Fuels (GNF) 10 x 10 fuel assembly is dropped over the reactor vessel, resulting in a total of 172 damaged rods in a GNF2 assembly and 169 damaged fuel rods in a GNF3 assembly. Accounting for partial length fuel rods with 560 fuel assemblies in the core, the damaged core fraction is 3.588×10^{-3} in a core loaded with GNF2 fuel and 3.417×10^{-3} in a core loaded with GNF3.

A radial peaking factor of 1.7 is applied to the fission product inventory of the damaged rods. The water above damaged fuel is not less than 23 feet in the reactor cavity. These values remain unchanged from the CLB. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The previously approved Alternate Source Term (AST) amendment takes credit for the decay of irradiated fuel in the two analyzed timeframes greater than 24 hours.

3.6.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged because of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water because of the accident.

As prescribed in RG 1.183 (Revision 0), the inventory of fission products available for release during the fuel handling accident is based on the maximum full power operation of the core with appropriate values for fuel enrichment, fuel burnup, and an assumed core power. The period of irradiation is of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory was determined using the appropriate isotope generation and depletion computer code referenced in RG 1.183 (Revision 0), ORIGEN-ARP. The licensee performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumes: (1) that the chemical form of radioiodine released from the fuel to the SFP consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, (2) the CsI released from the fuel completely dissociates in the pool water, and (3) because of the low pH of the pool water, the CsI re-evolves, and releases elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. FitzPatrick assumes that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously. These inputs and assumptions are consistent with the CLB.

As corrected by item 8 of Regulatory Issue Summary 2006-04 (ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read as follows:

> If the water depth above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85 percent) and organic iodine (0.15 percent) species results in the iodine above the water being composed of 57 percent elemental and 43 percent organic species.

Enclosure 1 of the licensee's supplement to the LAR dated August 31, 2023, contains calculation JAF-CALC-RAD-04410 (ML23243A946), in which Section 4.6 "Water Depth" describes the speciation of iodines used in the calculations. Appropriate values for iodine speciation are used in the licensee's RADTRAD calculations.

Consistent with the CLB, the licensee assumed that a minimum water level of 23 feet above the damaged fuel assembly is maintained for the bounding event of an assembly dropped over the reactor vessel. This minimum water covering acts as a barrier to many of the radionuclides released from the dropped assembly. Consistent with RG 1.183 guidance, the licensee assumed retention of all non-iodine particulates in the pool, while the iodine releases from the fuel gap into the pool are assumed to be decontaminated by an overall factor of 200. This decontamination factor results in 0.5 percent (i.e., 99.5 percent of the iodine is retained in the pool) of the radioiodine escaping the overlying water with a composition of 70 percent elemental iodine and 30 percent organic iodide. In accordance with RG 1.183, Appendix B, Regulatory Position 3, the licensee did not credit decontamination from water scrubbing for the noble gas constituents and assumed that 100 percent of the noble gas activity is released from the water covering the fuel.

This precondition is met for the reactor cavity, but not for the spent fuel pool, where the technical specification minimum water depth is 21 feet 7 inches. With this minimum water depth, a decontamination factor of 172.25 would be applied. FitzPatrick stated that the implied reduction in scrubbing efficiency is offset by the reduced number of fuel rods that are projected to be damaged by a fuel assembly drop over the spent fuel pool. The effective decontamination factor in RG 1.183 is based on an exponential function. In this function, more scrubbing occurs at the bottom of the water column than at the top of the water column. As such, a pool level of 21 feet 7 inches, a reduction of about 6 percent in pool depth, would result in a reduction in scrubbing efficiency of less than 6 percent. This is less than the 35 percent reduction in the amount of damaged rods in both GNF2 and GNF3 fuel assemblies, and hence the radionuclides released. This methodology for determining that the consequences of an accident over the spent fuel pool is bounded by the analyzed consequences of an accident over the reactor cavity is included in the CLB. The NRC staff finds the licensee's conclusion that the consequences of an FHA over the reactor cavity bounds those for an FHA over the spent fuel pool to be acceptable. This methodology was accepted in the safety evaluation associated with the June 7, 2002, FitzPatrick LAR, which changed the TS for handling irradiated fuel (ML022350228).

FitzPatrick's analysis of the source term for an FHA is consistent with the applicable guidance in Appendix B of RG 1.183, which identifies acceptable radiological analysis assumptions for a fuel handling accident. The NRC staff reviewed the licensee's provided source term and identified the corresponding values in the licensee's provided calculations and finds FitzPatrick's analysis acceptable.

3.6.1.1 Gap Release Fractions for AST

RG 1.183 provides guidance on AST implementation. Footnote 11 in Section 3.2 of RG 1.183 (Rev. 0), notes that the non-LOCA fuel rod gap fractions listed in Table 3 have been found acceptable for light-water reactor (LWR) fuel with peak burnup of 62 GWD/MTU, provided that the maximum linear heat generation rate (LHGR) does not exceed 6.3 kW/ft at burnups greater than 54 GWD/MTU. Section 4.1 "Source Term Assumptions" of calculation JAF-CALC-RAD-04410, in Enclosure 1 to the licensee's supplement dated August 31, 2023, addresses footnote 11 and states, "peak rod burnup of the fuel is less than 62 GWD/MTU" and "that the maximum linear heat generation rate does not exceed 6.3 kW/ft for any bundle with a peak rod burnup exceeding 54 GWD/MTU." The licensee evaluated the fission product release from the breached fuel following the guidance of RG 1.183 Regulatory Position 3.2, Table 3, which specifies the fraction of fission product inventory assumed to be present in the fuel rod gap. The assumptions used to determine the source term conform with guidance and are therefore acceptable.

3.6.2 Transport

3.6.2.1 Fuel Handling Accidents (FHA) over the Reactor Vessel in Secondary Containment

Releases from the FHA in secondary containment are via the plant vent stack. During normal operation, the system is in normal mode with two process radiation detectors monitoring the exhaust from the reactor building and the refueling floor zones exiting the plant vent stack. On alarm signal, the exhaust dampers close and the engineered-safeguard features (ESF (*e.g.,* primary/secondary containment, standby gas treatment, isolation capability)) are placed into service as the system is placed into emergency mode. In emergency mode, the ventilation automatically reconfigures and exhausts through ESF emergency filtration system charcoal and HEPA filters to remove halogens and particulates prior to discharging to the atmosphere via the plant vent. Consistent with RG 1.183, the analyzed FHA in the SFP involves a release over a two-hour period.

3.6.2.2 Fuel Handling Accidents (FHA) Atmospheric Dispersion values (χ/Q)

3.6.2.2.1 Atmospheric Dispersion

Onsite control room and offsite EAB and LPZ atmospheric dispersion values (χ/Qs) for FitzPatrick were previously approved by the NRC in the April 14, 2000, Safety Evaluation Report for Amendment No. 261 to Facility Operating License DPR-59, "Re: Changes to the Technical Specifications Regarding the Allowed Containment Leakage Rate" (ML18031A041).

Regulatory position 5.3 of RG 1.183 states that "Atmospheric dispersion factors (χ/Qs values) for the EAB, the LPZ, [and] the control room … that the staff approved during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified in this guide." In accordance with this guidance, the licensee states that the atmospheric dispersion values (χ/Qs) for the EAB, the LPZ, and the control room that were previously approved by the Staff are used in the FHA analysis.

The licensee, in this LAR, proposes new design basis accident χ/Qs for the Reactor Building (RB) vent release to the Technical Support Center (TSC). The determination of TSC χ/Q values were made using the ARCON96 atmospheric dispersion model (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," (ML17213A187)) pursuant to the guidance in RG 1.194. The NRC staff reviewed the licensee's new atmospheric dispersion analyses as described in sections 3.6.2.2.2 and 3.6.2.2.3 below.

3.6.2.2.2 Meteorology Data

The licensee provided supplemental information on August 31, 2023, regarding the atmospheric dispersion analysis described in the LAR. Hourly onsite meteorological data from calendar years 1985 through 1992 were used in the analysis. The meteorological data was formatted for the ARCON96 atmospheric dispersion code in order to calculate updated χ/Q values for the technical support center. This format contained hourly data on wind speed, wind direction, and atmospheric stability class.

The NRC staff previously completed a detailed review related to the acceptability and representativeness of the 1985 through 1992 onsite hourly meteorological data for the April 14, 2000, amendment regarding "Changes to the Technical Specifications Regarding the Allowed Containment Leakage Rate" (ML18031A041) and the September 12, 2002, amendment regarding the "Technical Specification Change to the Requirements for Handling Irradiated Fuel Assemblies" (ML022350228). Based on those reviews, the staff considers the onsite meteorological dataset from calendar years 1985 through 1992 suitable for use in making calculations for the atmospheric dispersion analyses used to support this LAR.

3.6.2.2.3 Technical Support Center Atmospheric Dispersion Analysis

The licensee used the computer code ARCON96 to estimate χ/Q values for the TSC for RB vent release of radioactive material. RG 1.194 endorses the ARCON96 model for determining χ/Q values to be used in the design basis evaluations of control room radiological habitability, and the NRC staff finds the licensee's use of ARCON96 acceptable for estimating χ/Q values for the TSC for RB vent release. The ARCON96 code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days.

Attachment A of the August 31, 2023, supplement includes a copy of the ARCON96 output file for the licensee's TSC χ/Q calculation. The file includes the meteorological data period, the lower and upper height measurements, and units of wind speed used in the analysis. The file also lists the source input values of vertical velocity, stack flow, stack radius, and diffusion coefficients. Also included are the release type, release height, building area for the release pathway as well as the distance to receptor, intake height, elevation difference, and direction to source for the release pathway. Finally, Attachment A lists the χ/Q values from the ARCON96 output for the 0-2, 2-8, 8-24, 24-96, and 96-720 hour time intervals for the TSC for the RB vent release.

The NRC staff confirmed the licensee's atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. Both the staff and licensee used a ground-level release assumption for the release pathway-receptor combination as well as the previously discussed source-receptor distance, direction, height, and area values. Based on the results of its confirmatory analysis, the staff finds the licensee's TSC χ/Q values acceptable for use in the radiological dose assessments.

3.6.3 Control Room (CR) Habitability for Fuel Handling Accidents (FHA)

Under accident conditions, habitability for the CR is provided by the CREVAS. This system provides habitability zone isolation and a positive pressure for the CR. In the fuel handling accident, the licensee credits the CREVAS for fuel handling accidents after 24 hours of fuel decay, up to 104 hours. After 104 hours the licensee's analysis shows that CREVAS is not required to meet regulatory requirements to maintain control room dose. Prior to 24 hours of decay time, the licensee relies on secondary containment and the SGTS to mitigate the release of radionuclides.

Prior to CREVAS actuation, the licensee evaluated the CR dose assuming a normal operation outside air intake of 2,112 cubic feet per minute (cfm). After CREVAS actuation, the licensee evaluated the CR dose assuming an emergency filtered air intake rate of 900 cfm. The licensee credited a CR removal efficiency of 97 percent for all iodine species. The licensee included a total of 400 cfm of unfiltered intake consisting of damper leakage of 300 cfm (including 10 cfm for ingress/egress) and 100 cfm of general CR air in-leakage. The technical specifications limit for unfiltered control room in-leakage is 100 cfm. This assumption provides significant margin to the timeframe from 24 to 104 hours when the CREVAS is relied upon to maintain control room dose within regulatory requirements.

The licensee evaluated the CR LOCA doses assuming continuous CR occupancy as defined in RG 1.183. RG 1.183, Section 4.2.6, defines continuous occupancy as follows:

> The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 days to 30 days. For the duration of the event, the licensee should assume the breathing rate of this individual to be 3.5×10^{-4} cubic meters per second.

Consistent with RG 1.183, regulatory position 4.2.1, the licensee considered all contributors to control room dose that will cause exposure to control room personnel other than infiltration of airborne contaminants into the control room. The other contributors include shine dose. The containment shine dose due to an FHA is insignificant compared to that due to a loss-of-coolant accident, or any accident where fuel damage to an operating core is assumed. Similarly, the external airborne cloud dose due to an FHA is insignificant. The analysis for these contributors is contained in Section 7.2 of the licensee's supplement dated August 31, 2023. Therefore, these two contributors to design basis accident dose can be eliminated from the analysis of the FHA. The FitzPatrick control room design has the CREVAS filtration behind and shielded by concrete block walls that are at least 2.5 feet thick. The NRC staff considers this shielding sufficient to eliminate separate consideration of the radiation shine from the filters in the dose analysis of the FitzPatrick FHA.

The NRC staff reviewed the LAR, its subsequent supplement, and the RAI response dated March 25, 2024. The RAI and the licensee's response clarified information associated with decay time and provided further information associated with the number of damaged fuel rods during an FHA. The staff determined that FitzPatrick used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The NRC staff finds that the inputs, analyses, and assumptions used in determining the hypothetical maximum exposed individual at each dose receptor location conform with applicable guidance and that the resulting TEDE dose values for the CR, EAB, and LPZ provided below by the licensee meet the applicable accident dose criteria in GDC 19, 10 CFR 50.67(b), and RG 1.183 and are, therefore, acceptable.

Post-FHA	**Post-FHA TEDE Dose (rem) 0–24 hrs post-accident Receptor Location**		
	Control Room	**EAB**	**LPZ**
GNF2 Fuel (<24 hrs Decay)	2.13E+00	2.98E+00	3.33E-01
GNF3 Fuel (<24 hrs Decay)	2.03E+00	2.84E+00	3.17E-01
Allowable TEDE Limit	**5.00E+00**	**6.30E+00**	**6.30E+00**

Post-FHA	**Post-FHA TEDE Dose (rem) 24–104 hrs post-accident Receptor Location**		
	Control Room	**EAB**	**LPZ**
GNF2 Fuel (24-104 hrs Decay)	4.95E+00	5.41E-01	6.05E-02
GNF3 Fuel (24-104 hrs Decay)	4.71E+00	5.15E-01	5.76E-02
Allowable TEDE Limit	**5.00E+00**	**6.30E+00**	**6.30E+00**

Post-FHA	Post-FHA TEDE Dose (rem) >104 hrs post-accident Receptor Location		
	Control Room	**EAB**	**LPZ**
GNF2 Fuel (>104 hrs Decay)	4.94E+00	2.85E-01	3.19E-02
GNF3 Fuel (>104 hrs Decay)	4.71E+00	2.71E-01	3.03E-02
Allowable TEDE Limit	**5.00E+00**	**6.30E+00**	**6.30E+00**

3.6.4 Assessment of the Technical Specifications Changes

A “recently irradiated” fuel is that fuel which has not been sufficiently decayed to allow relaxation of OPERABILITY requirements. Recently irradiated fuel can still be moved but the appropriate ESF systems need to be OPERABLE.

The licensee’s supplement, dated July 29, 2024 (ML24211A122), states:

> The term “recently” was added to the JAF TS and TS Bases from a 2002 TS amendment (Adams Accession Number ML022350228). With that amendment, the value of “96 hours” was only added to the TS bases. That amendment established 96 hours for all TSs as the time beyond which fuel is no longer considered “recently irradiated.” This amendment established different values for when fuel would be considered “recently irradiated” however the changes were only documented in the TS Bases which raised a concern by the NRC that the different TSs would use the same term, “recently irradiated” but the applicable time, per the TS Bases, would be different.

In response to the NRC staff’s concerns regarding the licensee’s original proposed changes to “Recently Irradiated Fuel Assemblies,” the licensee’s supplement dated July 29, 2024, provided proposed changes to revise the applicable TSs to be consistent with the changes made to the FHA analysis evaluated above. These changes involve a redefinition of recently irradiated fuel as it relates to SC and CREVAS System Instrumentation operability. For SC operability, the definition of recently irradiated fuel would be changed from the current 96 hours down to 24 hours. For CREVAS operability, the definition of recently irradiated fuel would be changed from the current 96 hours up to 104 hours. The changes were added in the APPICABILITY statements of the applicable LCOs affected by the redefinition of “recently irradiated” fuel.

TS LCO 3.0.1 states that LCOs shall be met when the unit is in the MODES or during other specified conditions of the Applicability statement of each Specification. The addition of the proposed changes in the Applicability statements of the affected LCOs would ensure that during movement of recently irradiated fuel assemblies in the secondary containment, the control room personnel are protected during a fuel handling event by entering applicable TS actions and implementing associated Required Actions.

The NRC staff's assessment, as described in this section, demonstrates that the modified changes would not adversely impact CREVAS's ability to function and perform as needed in providing assistance to maintain the control room operator doses within acceptable limits. Further, per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. Therefore, the NRC staff finds that the proposed changes to the TSs, as described in the licensee's supplement dated July 29, 2024, are acceptable because the TSs, as modified, will continue to meet 10 CFR 50.36(c)(2).

3.6.5 Conclusion

The NRC staff reviewed the guidance, assumptions, and methodology used by the licensee to assess the χ/Q values associated with postulated releases from the potential release pathway. The staff found that the licensee used methods consistent with regulatory guidance identified in Section 2.0 of this safety evaluation. The licensee used onsite meteorological data that complied with the guidance of RG 1.23. The inputs and assumptions used to calculate the Technical Support Center χ/Q values were also consistent with the guidance of RG 1.194. Therefore, on the basis of this review of the atmospheric dispersion analysis, the NRC staff finds the licensee's proposed χ/Q values acceptable for use in calculating the radiological dose assessments associated with the LAR.

The NRC staff also performed an independent review of all inputs, assumptions and initial conditions, including revised atmospheric dispersion coefficients in the licensee dose assessment files and performed independent confirmatory calculations using RADTRAD Version 5.0.3, as necessary, to ensure a thorough understanding of FitzPatrick's methods, and to verify that values used in the dose assessment code were in line with values provided in the LAR, its subsequent supplement, and the RAI responses. Based on the above, the NRC staff finds that the EAB, LPZ, and CR doses for the FHA meet the applicable accident dose criteria in GDC 19, 10 CFR 50.67(b), and RG 1.183 and are, therefore, acceptable. The staff also finds the proposed changes to the TSs are acceptable because the TSs, as modified, will continue to meet 10 CFR 50.36(c)(2).

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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 NSHC 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), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented as follows:

1. Do the proposed changes involve a significant increase in the probability, or consequences of an accident previously evaluated?

   Response: No

   The proposed amendment does not change any behavior or operation of the Fuel Handling Equipment due to transition from Refuel Bridge Mast NF-400 to NF-500 and time definition for Recently Irradiated Fuel.

The proposed change results in an increase in the Fuel Handling Accident (FHA) analysis radiological dose to a Control Room occupant. However, the resultant FHA Control Room dose consequences remain within the acceptance criteria provided by the NRC for use with the Alternative Source Term. These criteria are presented in 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed amendment does not result in a significant increase in the probability or consequences of any previously evaluated accident.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The NF-500 mast is similar enough in design and function to the NF-400 mast to not create the possibility of a new or different kind of accident. The proposed change does not significantly alter the Fuel Handling system design, create new failure modes, or change any modes of operation. Consequently, there are no new initiators that could result in a new or different kind of accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The change preserves the original design requirements and ensures adequate validation to confirm continued design capability. The margin of safety is considered to be that provided by meeting the applicable regulatory limits. The change to the FHA modeling results in an increase in Control Room dose following the FHA; however, since the Control Room dose following the design basis accident remains within the regulatory limits, there is not a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and based on this review, finds that the three standards of 10 CFR 50.92(c) are satisfied for the proposed changes to the FHA analysis and the time definition of "recently irradiated fuel" in the associated technical specifications. Also, no comments have been submitted on the proposed no significant hazards consideration determination. Therefore, the NRC staff makes a final finding that the license amendment request involves no significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on August 6, 2024. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (August 2, 2024; 89 FR 63226)]. Also, the NRC staff has made a final no significant hazards consideration finding, as discussed above. 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.

7.0 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: September 4, 2024

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT NO. 356 RE: UPDATE FUEL HANDLING ACCIDENT ANALYSIS (EPID L-2023-LLA-0109) DATED SEPTEMBER 4, 2024

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