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

July 19, 2018

Mr. Bryan C. Hanson
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. 319 RE: SECONDARY CONTAINMENT ACCESS OPENINGS (CAC NO. MG0239; EPID L-2017-LLA-0298)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 319 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 (TSs) in response to your application dated September 14, 2017, as supplemented by letter dated March 15, 2018.

The amendment modifies the TSs to allow for brief, inadvertent, simultaneous opening of both an inner and outer secondary containment personnel access doors during normal entry and exit conditions.

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

Sincerely,

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

Docket No. 50-333

Enclosures:
1. Amendment No. 319 to Renewed DPR-59
2. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 319
Renewed License No. DPR-59

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated September 14, 2017, as supplemented by letter dated March 15, 2018, 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. 319, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

James G. Danna, 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: July 19, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 319

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 page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.6.4.1-2	3.6.4.1-2

(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. 319, 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, ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	Initiate action to suspend OPDVRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	Verify one secondary containment access door in each access opening is closed, except when the access opening is being used for entry and exit.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.4	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.	In accordance with the Surveillance Frequency Control Program

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 319 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-59

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated September 14, 2017,[1] as supplemented by letter dated March 15, 2018,[2] Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for changes to the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) Technical Specifications (TSs). The proposed changes in the LAR would revise the requirements in TS 3.6.4.1, "Secondary Containment," associated with Surveillance Requirement (SR) 3.6.4.1.3. Specifically, SR 3.6.4.1.3 verifies that one secondary access door in each access opening is closed. The amendment would allow for brief, inadvertent, simultaneous opening of both an inner and outer secondary containment personnel access doors during normal entry and exit conditions.

The supplemental letter dated March 15, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 7, 2017 (82 FR 51650).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance that the NRC staff considered in its review of this LAR are described below.

2.1 Applicable Regulatory Requirements

The regulation under Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the regulatory requirements related to the contents of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following specific

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML17257A193.
[2] ADAMS Accession No. ML18074A049.

categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

Section 50.36(c)(2) to 10 CFR requires, in part, that LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility. When an LCO is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO is met.

Section 50.36(c)(3) to 10 CFR requires that TSs include SRs, which 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 LCOs will be met.

Section 50.67 to 10 CFR, "Accident source term," sets limits for the radiological consequences of a postulated design-basis accident (DBA) using an alternative source term (AST).

Section 50.67(b)(2) to 10 CFR Part 50 requires that the NRC may issue the amendment only if the applicant's analysis demonstrates 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 [Sievert] (25 rem [roentgen equivalent man])[3] 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) total effective dose equivalent (TEDE).

(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) total effective dose equivalent (TEDE) for the duration of the accident.

The regulations in Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants" (herein after referenced to as GDC) establish the minimum requirements for the principal design criteria of water-cooled nuclear power plants. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structure, systems, and components important to safety.

[3] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this Section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

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

- GDC 16, "Containment design," requires, in part, that the containment establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment.

- GDC 19, "Control room," requires, in part, that adequate radiation protection 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.

2.2 Applicable Guidance

The guidance documents the NRC staff considered in its review of this LAR are discussed below.

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

  - SRP Chapter 15, "Transient and Accident Analysis," Section 15.0.1, "Radiological Consequence Analyses using Alternative Source Terms," Revision 0, July 2000,[4] provides guidance to the NRC staff for the review of AST amendment requests. SRP Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

  - SRP Section 15.6.5, Appendix A, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident including Containment Leakage Contribution," Revision 1, July 1981.[5]

  - SRP Section 16.0, Revision 3, "Technical Specifications," March 2010.[6] As described therein, as part of the regulatory standardization effort, the NRC staff has prepared standard technical specifications (STS) for each of the light-water reactor nuclear designs. The STS contain guidance for the format and content of TSs that meet the requirements of 10 CFR 50.36. For this review, the NRC staff used NUREG-1433, Revision 4, "Standard Technical Specifications, General Electric BWR [Boiling Water Reactor]/4 Plants,"[7] and NUREG-1434, Revision 4, "Standard Technical Specifications, General Electric BWR/6 Plants,"[8] for guidance on the TS format. The STS for BWR/6 contains an exception that allows both doors in a secondary containment access opening to be open simultaneously for normal entry and exit.

---

[4] ADAMS Accession No. ML003734190.
[5] ADAMS Accession No. ML052350158.
[6] ADAMS Accession No. ML100351425.
[7] ADAMS Accession No. ML12104A192.
[8] ADAMS Accession No. ML12104A195.

- NUREG-1022, Revision 3, "Event Report Guidelines 10 CFR 50.72 and 10 CFR 50.73," January 2013,[9] discusses the reporting criteria contained in 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73 "Licensee event reporting system."

  Section 3.2.7 of NUREG-1022 discusses the reporting criteria in 10 CFR 50.72(b)(3)(v) and 10 CFR 50.73(a)(2)(v), which relate to events or conditions that could have prevented fulfilment of a safety function. This Section states, in part, that there are a limited number of single-train systems that perform a safety functions. For such systems, inoperability of a single train is reportable, even though the plant TSs may allow such a condition to exist for a limited time. This issue, as it relates to reportability for momentary inoperability of secondary containment, is discussed in the NRC staff's letter to Exelon dated January 8, 2015.[10]

- RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequence of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974.[11]

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000,[12] provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. The RG 1.183 provides guidance to licensees on acceptable application of AST (also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Nuclear Power Reactors," May 2003.[13]

2.3 Previous Amendments Regarding Radiological Consequences

License Amendment No. 276 dated September 12, 2002,[14] "James A. FitzPatrick Nuclear Power Plant - Amendment RE: Technical Specifications Change to the Requirements for Handling Irradiated Fuel Assemblies (TAC No. MB5328)," used an AST methodology for analyzing the radiological consequences of the fuel handling accident (also known as the refueling accident) using RG 1.183.

License Amendment No. 239 dated December 6, 1996,[15] "Issuance of Amendment for James A. FitzPatrick Nuclear Power Plant (TAC No. M92781)," approved the loss-of-coolant accident (LOCA) dose consequence analysis for FitzPatrick. The NRC staff also considered relevant information in the Updated Final Safety Analysis Report (UFSAR), which describes the design DBAs and evaluation of their radiological consequences for FitzPatrick.

---

[9] ADAMS Accession No. ML13032A220.
[10] ADAMS Accession No. ML14323A682.
[11] ADAMS Accession No. ML003739601.
[12] ADAMS Accession No. ML003716792.
[13] ADAMS Accession No. ML031490640.
[14] ADAMS Accession No. ML022350228.
[15] ADAMS Accession No. ML010960125.

3.0 TECHNICAL EVALUATION

3.1 Secondary Containment Safety Function

The secondary containment in a BWR Mark I design consists of a structure that completely encloses the primary containment and those components that may contain primary system fluid. The safety function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA to ensure the control room operator and offsite doses are within the regulatory limits. In conjunction with operation of the standby gas treatment system (SGTS), and closure of certain valves whose lines penetrate secondary containment, the secondary containment is designed to contain the fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment prior to release to the environment. For the secondary containment to be considered Operable, it must have adequate leak-tightness to ensure that the required vacuum can be established and maintained by one of the two redundant standby gas treatment (SGT) subsystems, when that subsystem is in operation. The secondary containment and SGTS together ensure radioactive material is contained.

The secondary containment boundary is the combination of walls, floor, roof, ducting, doors, hatches, penetrations, and equipment that physically form the secondary containment. A routinely used secondary containment access opening contains at least one inner and one outer door in an airlock configuration. In some cases, secondary containment access openings are shared such that there are multiple inner or outer doors. All secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit of personnel or equipment. The secondary containment serves as the containment during reactor refueling and maintenance operations and as an additional barrier when the primary containment is functional. There is no redundant train or system that can perform the secondary containment function should the secondary containment be inoperable.

3.2 Description of Proposed Technical Specification Changes

Limiting Condition for Operation 3.6.4.1 for FitzPatrick TS 3.6.4.1, "Secondary Containment," requires that the secondary containment be operable in Modes 1, 2, and 3, during movement of recently irradiated fuel assemblies in the secondary containment, and during operations with a potential for draining the reactor vessel. Surveillance Requirement 3.6.4.1.3 requires verification that one secondary containment access door in each access opening is closed, at a frequency in accordance with the Surveillance Frequency Control Program. The licensee is proposing to revise SR 3.6.4.1.3 as follows (changes shown in bold and italic): "Verify one secondary containment access door in each access opening is closed, ***except when the access opening is being used for entry and exit.***"

The licensee is proposing to revise SR 3.6.4.1.3 to allow for brief, inadvertent, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions. The licensee explained in the LAR that it is possible for an unintentional simultaneous opening of both the inner and outer secondary containment access doors during normal entry and exit. Based on the current wording in SR 3.6.4.1.3, a simultaneous opening of both inner and outer door in an access opening would require declaring the secondary containment inoperable. In addition, 10 CFR 50.72 and 10 CFR 50.73 require prompt notification and submittal of a licensee event report, regardless of the length of time of inoperability. The licensee stated that in a majority of cases, the secondary containment was restored to operable status in much less than the 4-hour action completion time for restoring the secondary containment. The licensee considers that declaring secondary containment inoperable for these brief occurrences is unwarranted, further stating that NUREG-1434 SR 3.6.4.1.3 contains an exception for both inner and outer doors in an access opening to be open for normal entry and exit. The proposed change would eliminate the need to declare the secondary containment inoperable for brief simultaneous opening of inner and outer access door and the attendant requirement to submit licensee event reports.

The LAR also provided TS Bases pages to be implemented with the associated TS change. These pages are provided for information only. Changes to the TS Bases are made in accordance with the TS Bases Control Program.

3.3 Secondary Containment Vacuum

In Section 3.0, "Technical Evaluation," of Attachment 1 to the LAR, the licensee stated that for "the secondary containment to be considered Operable, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained by a single SGT subsystem, when that subsystem is in operation." Currently, the FitzPatrick SR 3.6.4.1.4 requires verification that 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 cubic feet per minute (cfm). However, there are no SRs to verify that the required vacuum can be established within a specific draw down time. Therefore, in order for the NRC staff to complete its technical review of the licensee's proposed change, the NRC staff asked the licensee in request for additional information (RAI) letter dated January 29, 2018,[16] to discuss how the draw down time is periodically verified and to provide the results of the draw down tests performed during the past 3 years.

By letter dated March 15, 2018, the licensee stated in its RAI response that the LOCA analysis performed for FitzPatrick used RG 1.3. The licensee further stated that in the current licensing basis for FitzPatrick, it is an assumption that reactor building isolation and the start of the SGTS precludes significant fission product release. In the absence of draw down tests, the NRC staff review relied on the details provided in the LAR and the licensee's RAI response regarding the impact of inadvertent opening of the inner and outer door opening on the secondary containment vacuum.

The licensee stated that the time both doors may be open simultaneously will be limited to the time it takes to traverse through a door, typically less than 10 seconds. Technical Specification SR 3.6.4.1.1 requires verification of secondary containment vacuum is ≥ 0.25 inch vacuum water gauge. During normal conditions, this vacuum in secondary containment is maintained by the normal operating ventilation system. Empirical data from two previous licensee event reports dated November 6, 2015, and August 3, 2016, in which both secondary containment airlock doors were inadvertently opened simultaneously, demonstrate that this condition does not significantly impact the secondary containment differential pressure. These events did not result in the violation of the secondary containment vacuum requirement.

In its RAI response dated March 15, 2018, the licensee stated that recent performance testing of SGTS shows there is considerable margin to TS limits. The licensee stated that typical performance on an isolated reactor building is maintenance of building pressure at -1.35 ± 0.18 inches of water gauge below the outside pressure at a flow rate of

[16] ADAMS Accession No. ML18029A842.

5509 ± 224 cfm. Based on this information, the NRC staff concludes that the impact of the proposed change on SR 3.6.4.1.1, if any, would be minimal and acceptable.

3.4 Radiological Consequences

The NRC staff reviewed the impact of modifying FitzPatrick SR 3.6.4.1.3 to allow simultaneously opening of an inner and outer secondary containment personnel access door during normal entry and exit, on all DBAs currently analyzed in the FitzPatrick licensing basis that could have the potential for significant dose consequences. The FitzPatrick UFSAR Section 14.8.2, "Uprate Power Level Radiological Analyses," describes the DBAs and their radiological consequence analyses results.

The NRC staff assessed the proposed change against the licensee's design-basis radiological consequence dose analyses to ensure that it will not result in an increase in the radiological dose consequences and that the resulting calculated radiation doses will remain within FitzPatrick analyses of record. The NRC staff also reviewed the proposed change from the aspect of its impact on establishing and maintaining the secondary containment vacuum and preventing ground level exfiltration of radioactive materials.

3.4.1 *LOCA*

The FitzPatrick LOCA radiological dose consequence analysis was approved by letter dated December 6, 1996, as part of the power uprate application and is reflected in UFSAR Section 14.8.2.1.1, "Loss of Coolant Accident." The design-basis analysis assumptions are in accordance with RG 1.3. Regulatory Guide 1.3 and UFSAR Section 14.8.2.1.1 assume that all the noble gases and 25 percent of the halogens become airborne within the drywell at the time of the accident and are available for release. In addition, the FitzPatrick LOCA analysis assumes that all the noble gases and halogens leaking from the drywell into the reactor building are exhausted to the atmosphere via the SGTS and the main stack. Furthermore, the charcoal filter efficiency of the SGTS is 90 percent. This means that the FitzPatrick design-basis LOCA analysis assumes:

1. An instantaneous release at the time of the accident,
2. Secondary containment vacuum is establish at the start of the accident by the SGTS,
3. The secondary containment is maintained at the required vacuum for the duration of the accident,
4. The amount of release is reduced by the SGTS charcoal and particulate filters,
5. The release occurs from a pathway with an elevated release point (i.e., main stack), and
6. None of the release bypasses the secondary containment.

The simultaneous opening of both an inner and outer secondary containment door appears to allow a ground-level release to occur which bypasses the SGTS which is not consistent with the FitzPatrick licensing basis radiological dose consequence analysis for LOCA. Therefore, in order for the NRC staff to complete its technical review of the licensee's proposed change, the

NRC staff asked the licensee in its RAI letter dated January 29, 2018, to demonstrate that the proposed change does not bypass the SGTS and cause a ground-level release and that the functional capability of secondary containment is maintained, without an explicit secondary containment draw down time. In its RAI response dated March 15, 2018, the licensee stated, in part, that:

> Technical Evaluation EC 622932 was performed to evaluate the ability of the SGTS to maintain the Secondary Containment at sub-atmospheric pressure when the building is isolated, even if both doors in a single airlock are inadvertently opened at the same time. The evaluation demonstrates that when SGTS is able to maintain the building at 0.25 in. H_20 below the outside pressure at a flow rate of 6000 cfm (as required by TS); it will maintain the building approximately 0.004 in. H_20 below the outside pressure with both doors in an airlock open. This may also be understood qualitatively - for the SGTS fans to develop flow they require a source of suction. This source is from the Reactor Building. Therefore, flow must be into the building to permit outflow through the SGTS.
>
> The Technical Evaluation considered recent performance testing of SGTS and the Reactor Building and shows there is considerable margin to Technical Specification limits. Typical performance on an isolated Reactor Building is maintenance of building pressure at -1.35 ± 0.18 in. H_20 below the outside pressure at a flow rate of 5509 ± 224 cfm. Increased leak-tightness does not significantly affect the pressure at which building pressure will stabilize if both doors in an airlock are inadvertently opened, although it will shift the balance of in-flow from pre-existing leakage paths to [toward] the open airlock ([i.e.,] flow will be biased more to [from] the airlock as building leak-tightness improves).
>
> Therefore, the proposed change does not bypass the SGTS and cause a ground level release. The functional capability of Secondary Containment is maintained without an explicit Secondary Containment drawdown time.

The NRC staff finds, with reasonable assurance, that the proposed change to the TS will not result in an increase in any onsite or offsite dose because (1) the time both doors may be open simultaneously will be limited to the time it takes to traverse through a door, typically less than 10 seconds, and (2) the proposed change is consistent with the FitzPatrick licensing basis radiological dose consequence analysis for the design basis LOCA. Therefore, the NRC staff concludes that this change is acceptable with respect to the radiological consequences of the DBAs.

3.5 Refueling Accident

During normal operation, non-safety related systems are used to maintain the secondary containment at 0.25 inches of vacuum water gauge below the outside pressure to ensure that any leakage is into the building and that any secondary containment atmosphere exiting the building is via a monitored pathway. The refuel floor, which is inside the secondary containment, is maintained at a negative 0.25 inches of vacuum water gauge by normal operating ventilation systems. The refueling floor exhaust ductwork in the secondary containment is equipped with radiation monitors to detect a refueling accident. When a radiological release is sensed by the radiation monitors, a secondary containment isolation signal is generated. This initiates the SGTS and the normal ventilation system isolates. The radiation monitor is positioned such that it will detect the release and send a closure signal to the secondary containment isolation dampers.

Following a refueling accident, the secondary containment structure is maintained at a negative pressure by the SGTS ensuring that fission products released from the spent fuel pool to secondary containment can be collected and filtered prior to release to the environment. The refueling accident analysis assumes from 0 to 96 hours post-shutdown, that the secondary containment is operable and the initial release is through the reactor building vent. The reactor building exfiltration release is postulated to occur during the reactor building ventilation transition from normal mode to isolation mode. The brief vent release results from a delay in reactor building vent isolation that is slightly longer than the 10-second delay between radioactive materials entering the refuel floor exhaust duct, which initiates the refuel floor exhaust monitor isolation signal, and subsequently exiting the vent.

Conservatively, the DBA refueling accident radiological consequence analysis in UFSAR Section 14.8.2.1.4 assumes that following the start of a DBA refueling accident the secondary containment pressure of 0.25 inches of vacuum water gauge is credited at approximately 2 minutes. The license assumes that releases into the secondary containment prior to the 2-minute time leak directly to the environment as a ground level release with no filtration. After the assumed 2-minute time, these releases are filtered by the SGTS and released via the SGTS exhaust vent.

Based on its review, the NRC staff concludes that the licensee's DBA refueling accident analysis has sufficient conservatism by assuming a time of two minutes from the start of the DBA refueling accident. Margin exists to ensure that the secondary containment can be reestablished during brief, inadvertent, simultaneous opening of the inner and outer doors, and there is reasonable assurance that a failure of a safety system needed to control the release of radioactive material to the environment will not result. The brief, inadvertent, simultaneous opening of the secondary containment access doors does not impact the design bases and will not result in an increase in any on-site or off-site dose.

3.6 Administrative Controls

In Section 3.0 of Enclosure 1 of the LAR, the licensee stated, in part, that:

> Personnel are trained in Nuclear General Employee Training (NEGT) to not open secondary containment personnel access door if the indicating light is illuminated. Additionally, administrative controls exist specifying that the user verifies an indicating light and pauses for 5 seconds prior to proceeding. The intent of these administrative controls is to allow personnel [who] may have entered the airlock at an earlier time to successfully traverse to the exit door prior to the next attempted entry/exit.

The LAR also stated:

> The proposed change does not involve planned simultaneous opening of redundant secondary containment personnel access doors. For situations that involve planned simultaneous opening of the doors, secondary containment will be declared inoperable and the appropriate TS action will be followed.

The statements assure that administrative controls will continue to be maintained and that the proposed change will not be used for planned simultaneous opening of the access doors.

3.7 Evaluation of TS Changes

The NRC staff reviewed the proposed changes to the TSs by considering whether the proposed SR would continue to meet the requirements of 10 CFR 50.36. The regulations do not specify the format or content of the individual specifications. The proposed changes to SR 3.6.4.1.3 would add an exception to the applicability of the SR by allowing both doors in a secondary containment access opening to be opened simultaneously for normal entry and exit. The change clarifies the applicability of the requirement, but does not change the method of verifying secondary containment integrity. The NRC staff determined that the proposed SR would continue to meet the requirements in 10 CFR 50.36(c)(3), which specifies the SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, facility operation will be within safety limits, and the LCOs will be met.

The TSs for FitzPatrick are based on the improved STS. The NRC staff reviewed the format and content of the corresponding TSs in NUREG-1433, Revision 4, and NUREG-1434, Revision 4, to determine if the proposed change is consistent with the format and content of the NUREGs. The NRC staff found that the proposed changes are consistent with the format of NUREG-1433 and NUREG-1434 and the content of NUREG-1434. The corresponding TS in NUREG-1434 has a similar SR to the proposed revised SR for FitzPatrick. The licensee's letter dated September 14, 2017, provided revised TS Bases pages to be implemented with the associated TS changes. These pages were provided for information only and will be revised by the licensee in accordance with the TS Bases Control Program.

3.8 Technical Evaluation Conclusion

Based on its evaluation, the NRC staff concludes that the proposed changes are acceptable. The NRC staff has reasonable assurance that with the proposed change, the functional capability of secondary containment for FitzPatrick will be maintained. The NRC staff finds, with reasonable assurance, that the proposed change to the FitzPatrick TSs will continue to comply with the regulatory requirements and guidance identified in Section 2.0 above. In addition, the NRC staff finds that the proposed changes are acceptable with regard to the radiological consequences of the postulated DBAs.

The NRC staff review was limited to the licensee's request to provide an allowance for the brief, inadvertent, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions. Planned activities that could result in the simultaneous opening of redundant secondary containment access openings, such as maintenance of a secondary containment personnel access door or movement of large equipment through the opening that would take longer than the normal transit time, were considered outside the scope of the staff's review.

4.0 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 25, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. 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 (82 FR 51650). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: K. Bucholtz, NRR
N. Karipineni, NRR

Date: July 19, 2018

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 319 RE: SECONDARY CONTAINMENT ACCESS OPENINGS (CAC NO. MG0239; EPID L-2017-LLA-0298) DATED JULY 19, 2018

DISTRIBUTION:
Public
PM File Copy
RidsACRS_MailCTR
RidsNrrDorlLpl1
RidsNrrDraArcb
RidsNrrDssScpb
RidsNrrDssStsb
RidsNrrLALRonewicz
RidsNrrLAJBurkhardt
RidsNrrPMFitzPatrick
RidsRgn1MailCenter
KBucholtz, NRR
MChernoff, NRR
MGray, R1

ADAMS Accession No.: ML18180A372 *via email ** via memo

OFFICE	DORL/LPL1/PM	DORL/LPL1/LA	DRA/ARCB/BC*	DSS/SCPB/BC**
NAME	THood	LRonewicz (JBurkhardt for)	KHsueh	RDennig
DATE	07/02/2018	07/06/2018	04/26/2018	06/01/2018
OFFICE	DSS/STSB/BC	OGC – NLO*	DORL/LPL1/BC	DORL/LPL1/PM
NAME	VCusumano	BHarris	JDanna	THood
DATE	07/10/2018	07/12/2018	07/18/2018	07/19/2018

OFFICIAL RECORD COPY