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

July 21, 2020

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. 338 RE: ALTERNATIVE SOURCE TERM FOR CALCULATING LOSS-OF-COOLANT ACCIDENT DOSE CONSEQUENCES (EPID L-2019-LLA-0171)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 338 to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) in response to your application dated August 8, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19220A043), as supplemented by letters dated August 27, 2019; January 16, 2020; and March 30, 2020 (ADAMS Accession Nos. ML19261A168, ML20017A052, and ML20090E279, respectively), that requested adoption of the alternative source term in accordance with Title 10 of the *Code of Federal Regulations* Section 50.67 for use in calculating the loss-of-coolant accident dose consequences at FitzPatrick.

The license amendment request for use of an alternative source term in evaluating the consequences of the loss-of-coolant accident is a full implementation of the alternative source term as provided in Section 1.2.1 of Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000.

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

In conjunction with the license amendment request, your letter dated August 8, 2019, also contained a request for an exemption from the requirements of Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to Title 10 of the *Code of*

*Federal Regulations* Part 50, dependent on the approval of alternative source term. The NRC staff has approved your exemption request separately from this amendment request.

Sincerely,

***/RA/***

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

Docket No. 50-333

Enclosures:
1. Amendment No. 338 to DPR-59
2. Safety Evaluation

cc: Listserv

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

EXELON FITZPATRICK, LLC

AND

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 338
Renewed Facility Operating License No. DPR-59

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

   A. The application for amendment by Exelon FitzPatrick, LLC and Exelon Generation Company, LLC (collectively, the licensees) dated August 8, 2019, as supplemented by letters dated August 27, 2019; January 16, 2020; and March 30, 2020, 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. 338, 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 75 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2020.07.21 10:29:08 -04'00'

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

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

Date of Issuance: July 21, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 338

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 a marginal line indicating the area of change.

| Remove Page | Insert Page |
|---|---|
| Page 3 | Page 3 |

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

| Remove Pages | Insert Pages |
|---|---|
| 3.1.7-1 | 3.1.7-1 |
| 3.3.6.1-10 | 3.3.6.1-10 |
| 3.6.1.3-9 | 3.6.1.3-9 |
| 3.6.1.8-1 | 3.6.1.8-1 |
| 3.6.1.8-2 | --- |
| 3.6.4.1-2 | 3.6.4.1-2 |
| 5.5-8 | 5.5-8 |
| 5.5-9 | 5.5-9 |

(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. 338, 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,

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS PER TRIP SYSTEM | CONDITIONS REFERENCED FROM REQUIRED ACTION C.1 | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 5. Reactor Water Cleanup (RWCU) System Isolation | | | | | |
| a. RWCU Suction Line Penetration Area Temperature — High | 1, 2, 3 | 1 | F | SR 3.3.6.1.3<br>SR 3.3.6.1.7 | ≤ 144°F |
| b. RWCU Pump Area Temperature — High | 1, 2, 3 | 1 per room | F | SR 3.3.6.1.3<br>SR 3.3.6.1.7 | ≤ 165°F for Pump Room A and ≤ 175°F for Pump Room B |
| c. RWCU Heat Exchanger Room Area Temperature — High | 1, 2, 3 | 1 | F | SR 3.3.6.1.3<br>SR 3.3.6.1.7 | ≤ 155°F |
| d. SLC System Initiation | 1, 2, 3 | 2(d) | I | SR 3.3.6.1.7 | NA |
| e. Reactor Vessel Water Level — Low (Level 3) | 1, 2, 3 | 2 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.5<br>SR 3.3.6.1.7 | ≥ 177 inches |
| f. Drywell Pressure — High | 1, 2, 3 | 2 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.5<br>SR 3.3.6.1.7 | ≤ 2.7 psig |
| 6. Shutdown Cooling System Isolation | | | | | |
| a. Reactor Pressure — High | 1, 2, 3 | 1 | F | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.5<br>SR 3.3.6.1.7 | ≤ 74 psig |
| b. Reactor Vessel Water Level — Low (Level 3) | 3 | 2 | J | SR 3.3.6.1.1<br>SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>SR 3.3.6.1.5<br>SR 3.3.6.1.7 | ≥ 177 inches |

(continued)

(d) SLC System Initiation only inputs into one of the two trip systems and only isolates one valve in the RWCU suction and return line.

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.1.3.7 | Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal. | In accordance with the Surveillance Frequency Control Program |
| SR 3.6.1.3.8 | Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break. | In accordance with the Inservice Testing Program |
| SR 3.6.1.3.9 | Remove and test the explosive squib from each shear isolation valve of the TIP System. | In accordance with the Surveillance Frequency Control Program |
| SR 3.6.1.3.10 | Verify combined main steam line leakage rate is ≤ 200 scfh, and ≤ 100 scfh for any one steam line, when tested at ≥ 25 psig. | In accordance with the Primary Containment Leakage Rate Testing Program |
| SR 3.6.1.3.11 | Verify the leakage rate of each air operated testable check valve associated with the LPCI and CS Systems vessel injection penetrations is within limits. | In accordance with the Primary Containment Leakage Rate Testing Program |

## 3.6 CONTAINMENT SYSTEMS

### 3.6.1.8 Deleted

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.6.4.1.1 | ----------NOTE----------<br>Not required to be met for 4 hours if analysis demonstrates that one Standby Gas Treatment (SGT) subsystem is capable of establishing the required secondary containment vacuum.<br>----------<br><br>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 |

## 5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

a. Demonstrate for each of the Engineered Safeguards systems that an inplace test of the HEPA filters shows a penetration and system bypass less than the value specified below when tested in accordance with Sections C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flowrate specified below.

| Engineered Safeguards Ventilation System | Penetration and System Bypass | Flowrate (scfm) |
|---|---|---|
| Standby Gas Treatment System | 1.5% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | 1.5% | 900 to 1,100 |

b. Demonstrate for each of the Engineered Safeguards systems that an inplace test of the charcoal adsorber shows a penetration and system bypass less than the value specified below when tested in accordance with Sections C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ASME N510-1980 at the system flowrate specified below.

| Engineered Safeguards Ventilation System | Penetration and System Bypass | Flowrate (scfm) |
|---|---|---|
| Standby Gas Treatment System | 1.0% | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | 0.5% | 900 to 1,100 |

(continued)

## 5.5 Programs and Manuals

5.5.8 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the Engineered Safeguards systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Section C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in series, in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C (86°F) and the relative humidity specified below.

| Engineered Safeguards Ventilation System | Penetration | RH |
|---|---|---|
| Standby Gas Treatment System | 1.5% | ≥ 70% |
| Control Room Emergency Ventilation Air Supply System | 1.5% | ≥ 95% |

d. Demonstrate for each of the Engineered Safeguards systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

| Engineered Safeguards Ventilation System | Delta P (inches wg) | Flowrate (scfm) |
|---|---|---|
| Standby Gas Treatment System | 5.7 | 5,400 to 6,600 |
| Control Room Emergency Ventilation Air Supply System | 5.8 | 900 to 1,100 |

e. Demonstrate that the heaters for the Standby Gas Treatment System dissipate > 29 kW when tested in accordance with ASME N510-1975.

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 338

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

EXELON FITZPATRICK, LLC

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

## 1.0 INTRODUCTION

By letter dated August 8, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19220A043), as supplemented by letters dated August 27, 2019; January 16, 2020; and March 30, 2020 (ADAMS Accession Nos. ML19261A168, ML20017A052, and ML20090E279, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) Technical Specifications (TSs). The supplemental letters dated August 27, 2019; January 16, 2020; and March 30, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed in the *Federal Register* on November 19, 2019 (84 FR 63899), and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination.

The proposed changes would revise the radiological assessment calculational methodology for the design-basis accident (DBA) loss-of-coolant accident (LOCA) at FitzPatrick through application of the alternative source term (AST). The license amendment request (LAR) for the LOCA is a full implementation of the AST, as provided in Section 1.2.1 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792).

## 2.0 REGULATORY EVALUATION

### 2.1 System Descriptions

#### 2.1.1 Primary Containment Isolation Valves

The function of the primary containment isolation valves (PCIVs) is to limit fission product release during and following postulated DBAs to within limits. Automatic closure of the PCIVs

within applicable closure times, in combination with accident mitigation systems, ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the DBA dose consequence analysis. Based on TS 5.5.6, "Primary Containment Leakage Rate Testing Program":

- The peak primary containment internal pressure for the design-basis LOCA, $P_a$, is 45 pounds per square inch gauge (psig).

- The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, shall be 1.5 percent of containment air weight per day.

- The primary containment leakage rate acceptance criterion is 1.0 $L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are 0.60 $L_a$ for the Type B and Type C tests, and 0.75 $L_a$ for the Type A tests.

Based on the current TSs, main steam isolation valve (MSIV) leakage is accounted for in the leakage acceptance criterion of 0.75 $L_a$ for Type A test and 0.6 $L_a$ for Type B and Type C tests. The exemption request and the AST dose consequence analysis with its associated TS changes, as proposed in the LAR would allow the licensee to exclude the MSIV leakage from the acceptance criterion. The NRC staff has evaluated the licensee's exemption request separately from this LAR.

Surveillance Requirement (SR) 3.6.1.3.10 addresses the periodic surveillance of leakage limits and the surveillance frequency of the MSIVs. The allowed MSIV leakage rate and the leakage path are inputs to the AST dose consequence analysis.

2.1.2 Main Steam Isolation Valves

The four main steam lines (MSLs), which penetrate the drywell, are automatically isolated by the MSIVs. There are two MSIVs on each steam line – one inside containment and one outside containment. The MSIVs are functionally part of the primary containment boundary, and leakage through these valves provides a potential leakage path for fission products. In the current design basis, the leakage through the MSIVs goes to the turbine stop valves (TSVs), condenser, and ultimately to the environment as a ground level release.

2.1.3 Main Steam Leakage Collection System

The main steam leakage collection (MSLC) system supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after an accident. The MSLC system consists of two independent and redundant subsystems. Each subsystem collects leakage from the stem packing and the seats of all four outboard MSIVs. Each subsystem consists of valves, controls, and piping that can be aligned to direct the leakage from the MSIVs to the standby gas treatment system (SGTS) for processing. Leakage from the MSIV stem packing and seats (i.e., through-wall) to the reactor building (RB) would go through the SGTS if it is in service. Otherwise, it goes through the RB heating, ventilation, and air-conditioning system to the environment.

### 2.1.4 Standby Gas Treatment System and Control Room Emergency Ventilation Air Supply System

The secondary containment is a structure that encloses the primary containment, including 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 (CR) operator and offsite doses are within the regulatory limits. There is no redundant train or system that can perform the secondary containment function, should the secondary containment be inoperable.

The secondary containment boundary is the combination of walls, floor, roof, ducting, doors, hatches, penetrations, and equipment that physically forms the secondary containment. Secondary containment operability is based on its ability to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. To prevent ground-level exfiltration of radioactive material while allowing the secondary containment to be designed as a mostly conventional structure, the secondary containment requires support systems to maintain the pressure at less than atmospheric pressure. During normal operation, non-safety-related systems are used to maintain the secondary containment at a slight negative pressure to ensure any leakage is into the building and that any secondary containment atmosphere exiting is by a pathway monitored for radioactive material. However, during normal operation, it is possible for the secondary containment vacuum to be momentarily less than the required vacuum for several reasons, such as during wind gusts or swapping of the normal ventilation subsystems.

During emergency conditions, the SGTS is designed to be capable of drawing down the secondary containment to a required vacuum within a prescribed time and to continue to maintain the negative pressure as assumed in the accident analysis. The leaktightness of the secondary containment, together with the SGTS, ensure that radioactive material is either contained in the secondary containment or filtered through the SGTS filter trains consisting of high-efficiency particulate air (HEPA) and charcoal absorbers before being discharged to the outside environment by an elevated release point.

The CR emergency ventilation air supply system provides outside air into the CR envelope through an emergency filter train consisting of HEPA filters and charcoal absorbers.

## 2.2 Proposed Changes

The primary purpose of the amendment request is to revise the current dose consequence analysis for FitzPatrick. A copy of the AST LOCA dose calculation was provided in Attachment 6 to the LAR. In accordance with the AST LOCA analysis assumptions, revisions to the FitzPatrick TSs are proposed to ensure that the assumptions in the analysis are consistent with the TSs.

The proposed TS changes would do the following:

- Revise SR 3.6.1.3.10 to increase the combined MSIV leakage rate limit for all four steam lines from 46 standard cubic feet per hour (scfh) to 200 scfh when tested at greater than or equal to 25 psig and add a leakage limit of less than or equal to 100 scfh for a single main steam line.

Revised SR 3.6.1.3.10 would state (deletions shown and additions in bold and underlined):

> Verify combined main steam line leakage rate is ≤ ~~46~~**200** scfh**, and ≤ 100 scfh for any one steam line,** when tested at ≥ 25 psig.

- Delete TS 3.6.1.8, "Main Steam Leakage Collection (MSLC) System," and the associated Bases in their entirety and revise the TSs Table of Contents to reflect the deletion.

- Revise TS 3.1.7, "Standby Liquid Control (SLC) System," and associated instrumentation in TS 3.3.6.1, "Primary Containment Isolation Instrumentation."

- Revise the ventilation filter testing program (VFTP) – TS 5.5.8.a and TS 5.5.8.c.

- Adopt Technical Specifications Task Force (TSTF) Traveler TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements" (ADAMS Accession No. ML17267A226), which revises TS 3.6.4.1, "Secondary Containment," SR 3.6.4.1.1. This proposed change would allow the secondary containment vacuum limit to not be met, provided that the SGTS remains capable of establishing the required secondary containment vacuum.

## 2.3 Reasons for the Proposed Change

The licensee provided the following reasons for the requested changes:

- Incorporate a revised source term based on the core average exposure. The core average source term allows increased operational flexibility by bounding a range of core average exposures and fuel enrichments.

- Increase the allowable MSIV leakage to significantly reduce the amount of rework on the MSIVs. Refurbishment of an MSIV to meet the current leakage limit in SR 3.6.1.3.10 is a labor-intensive effort, which results in cumulative worker radiation dose and expenditure of resources. The proposed change would lower personal radiation exposure and save costs.

- Remove the requirement for post-accident operation of the MSLC system.

## 2.4 Applicable Regulatory Requirements and Guidance

### 2.4.1 Radiological Consequences

The NRC staff evaluated the radiological consequences of affected DBAs for implementation of the AST methodology at FitzPatrick, as proposed by the licensee, against the dose criteria specified in 10 CFR 50.67(b)(2). These criteria are 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE for access and occupancy of the CR for the duration of the postulated fission product release.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183; NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Standard Review Plan or SRP), Section 15.0.1; and Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 (GDC) 19. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

The following NRC requirements and guidance documents are applicable to the NRC staff's review of the LAR.

- 10 CFR Section 50.67, "Accident source term"
- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, GDC 19, "Control Room"
- RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000
- Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347)
- J. E. Cline & Associates, Inc., "MSIV Leakage Iodine Transport Analysis," dated March 26, 1991 (ADAMS Accession No. ML003683718) [Reference A-9 in RG 1.183, Revision 0]
- RG 1.196, Revision 1, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated January 2007 (ADAMS Accession No. ML063560144)
- NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," dated October 1978
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Revision 3, dated March 2007
- NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," dated March 2007
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000
- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report" dated February 1995 (ADAMS Accession No. ML041040063)
- NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," dated June 1993 (ADAMS Accession No. ML063480542)

- J. Schaperow, et al., "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," U.S. Nuclear Regulatory Commission, AEB 98-03, dated December 9, 1998 (ADAMS Accession No. ML011230531)

- Staff Requirements Memorandum (SRM) to Commission Papers (SECY)-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," dated July 2, 2019 (ADAMS Accession No. ML19183A408)

- NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," dated June 3, 1999

2.4.2 Emergency Core Cooling

- GDC 35, "Emergency core cooling": Insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any loss of coolant at a rate so that fuel and clad damage that could interfere with continued effective core cooling will be prevented.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which, in part, establishes standards for the calculation of emergency core cooling system (ECCS) accident performance and acceptance criteria for that calculated performance. In particular, 10 CFR 50.46(b)(4), "Coolable geometry," requires that calculated changes in core geometry shall be such that the core remains amenable to cooling.

2.4.3 Atmospheric Dispersion

The LAR provides selected atmospheric dispersion ($\chi/Q$) values (which are also known as relative concentrations) for use in radiological dose analysis. The NRC staff's evaluation of the proposed $\chi/Q$ values is based upon the following regulations, regulatory guides, and guidance documents:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control Room," with respect to the atmospheric dispersion used in the radiological dose analysis to demonstrate compliance with dose limits inside the CR during radiological accident conditions

- NUREG-0800, SRP Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," which provides guidance regarding calculating atmospheric dispersion factors for postulated accidental airborne radioactive releases

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," which includes guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing and operation

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," which discusses the need for an evaluation of the radiological consequences of DBAs in the CR

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” which discusses acceptable approaches for estimating short-term (i.e., 2 hours to 30 days after an accident) average χ/Q values near the buildings at CR ventilation air intakes and at other locations of significant airborne air in-leakage to the CR envelope caused by postulated DBA accident radiological releases

2.4.4 Environmental Qualification of Electric Equipment

Section 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” of 10 CFR identifies requirements for establishing a program for qualifying electric equipment that is important to safety as defined in 10 CFR 50.49(b).
Section 50.49(e)(1) of 10 CFR requires that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe DBA during and following which this equipment is required to remain functional.
Section 50.49(e)(2) of 10 CFR requires that humidity during DBAs must be considered.
Section 50.49(e)(4) of 10 CFR requires that the radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe DBA during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects. Section 50.49(b)(2) of 10 CFR requires qualification of non-safety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety-related equipment. Section 50.67 of 10 CFR provides an optional provision for licensees to revise the AST used in design-basis radiological analyses.

RG 1.183 states that licensees may use either the AST or Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites” (ADAMS Accession No. ML021720780), assumptions for performing the required environmental qualification (EQ) analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID-14844) on EQ doses.

2.4.5 Technical Specifications

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

The regulation at 10 CFR 50.36(b) requires that each license authorizing reactor operation include TSs derived from the analyses and evaluation included in the safety analysis report and amendments thereto.

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)(i), 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. 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.

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement), which discussed the criteria to determine the items that are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Section 50.36(c)(3) of 10 CFR requires TSs to include items in the category of 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 50.36(c)(5) of 10 CFR requires TSs to include items in the category of administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The NRC staff's guidance for review of TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," dated March 2010 (ADAMS Accession No. ML100351425).

## 3.0 TECHNICAL EVALUATION

### 3.1 Dose Consequences Using Alternative Source Term

#### Background

Since the publication of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future

light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to reanalyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design-basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and RG 1.183. The NRC's traditional methods for calculating the radiological consequences of DBAs are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the AST methodology and with the TEDE criteria provided in 10 CFR 50.67. RG 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design-basis radiological analyses using an AST. The guidance in RG 1.183 supersedes corresponding radiological analysis assumptions provided in other RGs and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

Discussion

The licensee performed a full implementation of the AST as defined in RG 1.183, Section 1.2.1, which states that, "At a minimum for full implementations, the DBA LOCA must be re-analyzed using the guidance in Appendix A of this guide." The licensee has determined that the current TID-14844 accident source term will remain the licensing basis for equipment qualification, NUREG-0737 evaluations other than for CR habitability, and for the radiological consequence analyses of non-LOCA accidents described in the FitzPatrick Updated Final Safety Analysis Report (UFSAR).

The licensee performed a review of its shielding study developed in response to NUREG-0737, Item 11.B.2, and evaluated radiation exposure to plant personnel performing vital missions in support of accident mitigation and safe shutdown following a LOCA. The source terms were based on traditional TID-14844 assumptions. The nuclide release fractions and release timing specified in the AST methodology outlined in RG 1.183 differ from those outlined in TID-14844 and NUREG-0737. The difference in the release fractions has the potential to affect the dose rates in vital areas where piping containing post-LOCA sump fluid is located. NUREG-0933, Generic Issue 187, reported that exposure to containment atmosphere sources based on traditional TID-14844 source term and AST methodology produced similar integrated doses and that the integrated AST doses from exposure to post-LOCA sump fluid do not exceed those based on TID-14844 assumptions until 145 days after an event at a BWR. This is due to the difference in the assumption regarding the release fraction for cesium (30 percent (%) in NUREG-1465 vs. 1 percent in TID-14844). Based on NUREG-0933, the licensee concluded, and the NRC staff agreed, that the differences in the release fractions associated with AST methodology would have little impact on the local dose rates during the 30-day post-LOCA mission time. Since the local dose rates are not expected to be significantly impacted by the AST during the first 30 days following a LOCA, the conclusions of the shielding study with

respect to operator exposure would not significantly change by expressing the mission doses in terms of TEDE.

The licensee performed dose calculations at the EAB for the worst 2-hour period following the onset of the accident. The integrated doses at the outer boundary of the LPZ and the integrated dose in the FitzPatrick CR were evaluated for the duration of the accident. The licensee used the computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03 (RADTRAD), as described in NUREG/CR-6604 to evaluate the FitzPatrick LOCA dose consequences. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performed independent confirmatory dose evaluations, as needed, using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 of the appendix to this SE. The atmospheric dispersion factors used by the licensee are shown in Tables 2 and 3 of the appendix of this SE. The data and assumptions used in the LOCA analysis are provided in Tables 4 and 5 of the appendix to this SE.

The licensee generated the core radionuclide inventory for use in determining source term releases using ORIGEN-ARP-based methodology. The inventory, consisting of 60 dose significant isotopes at the end of fuel cycle curie levels, formed the input for the RADTRAD dose evaluation code.

As stated in RG 1.183, the release fractions associated with the light-water reactor core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved light-water reactor fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatts per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU. The licensee used a power level of 2,586.7 megawatts thermal (MWt) to account for 2 percent measurement uncertainty (2,536 x 1.02 = 2586.7). The licensee calculated the accident fission product inventory based on an average core burnup consisting of burnups ranging from 12,000 to 43,000 MWD/MTU.

The licensee used committed effective dose equivalent and effective dose equivalent dose conversion factors from Federal Guidance Reports (FGR) 11[1] and 12[2] to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and dose conversion factors from FGR-11 and FGR-12 is consistent with RG 1.183 guidance and is, therefore, acceptable to the NRC staff.

### 3.1.1 Loss-of-Coolant Accident

The radiological consequence design-basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary reactor coolant system (RCS) piping. As applied in nuclear technology, the deterministic approach generally deals with evaluating the

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[1] K. F. Eckerman et al., "Limiting Values of Radionuclide Intakes and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988

[2] K. F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993

safety of a nuclear power plant in terms of the consequences of a predetermined bounding subset of accident sequences. The LOCA accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling, which results in a significant amount of reactor core fuel damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence but is representative of a class of severe damage incidents that was evaluated in the development of RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis transient analyses.

When using the AST for the evaluation of a design-basis LOCA, it is assumed that the initial fission product release to the containment will last 2 minutes and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 2 minutes, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends, and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.5 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

In the evaluation of the design-basis LOCA radiological analysis, the licensee included dose contributions from the following activity release pathways:

- primary containment leakage to the RB
- primary containment bypass leakage directly to the environment
- engineered safety feature (ESF) leakage to the RB
- MSIV leakage to the environment

The licensee included the following DBA LOCA dose contributors to the CR analysis:

- contamination of the CR atmosphere by the filtered intake of radioactive material contained in the radioactive plume released from the facility

- contamination of the CR atmosphere by the unfiltered infiltration of airborne radioactive material from areas and structures adjacent to the CR envelope

- radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud)

- radiation shine from radioactive material in the RB

- radiation shine from radioactive material in systems and components external to the CR envelope (e.g., radioactive material buildup on ventilation filters)

3.1.1.1 Assumptions on Transport in Primary Containment

3.1.1.1.1 Containment Leak Rate

The design-basis LOCA considered in this evaluation is a complete and instantaneous circumferential severance of one of the recirculation loops, which would result in the maximum

fuel temperature and primary containment pressure among the full range of LOCA scenarios. The pipe break results in a blow-down of the reactor pressure vessel (RPV) liquid and steam to the drywell by the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases through a vent system into the suppression pool water, thereby condensing the steam and reducing the drywell pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

FitzPatrick has a BWR-Mark I containment. The FitzPatrick Mark I primary containment consists of two compartments. The two compartments are connected by a vent system that allows steam released from the reactor vessel (located in the drywell) to flow into the suppression pool. The primary containment leakage is limited by TS 5.5.6.b to 1.5 volume percent per day at the peak drywell pressure of 45 psig. Because of post-accident containment depressurization, this leakage rate will decrease with time. As described in RG 1.183, Appendix A, Section 3.7, for BWRs, leakage may be reduced after the first 24 hours if supported by plant configuration and analyses to a value not less than 50 percent of the TS leak rate. In the design-basis LOCA analysis, the licensee assumes that the TS primary containment leak rate is reduced by a factor of 2 after 24 hours.

3.1.1.1.2 Containment Mixing

The design-basis LOCA for dose consequence analyses assumes that the mass and energy (steaming and steam condensation) created by reflooding (arresting RPV failure) and core quenching will provide enough energy to mix the drywell and wetwell air when vacuum breaker cycling occurs during this pressure transient. The licensee assumed that the radioactivity release is diluted into the larger volume of the wetwell plus drywell air spaces after 2 hours. Before this time, the radioactivity is only assumed to be released into the drywell net free volume. The NRC staff expects that a significant percentage of the fission products (other than noble gases and iodine in organic form) in the drywell air transferred to the wetwell air space will be scrubbed by the suppression pool water. Conservatively, the licensee did not credit any reduction in fission products transferred to the suppression pool air space from the drywell by suppression pool scrubbing. Instead, the licensee assumed a well-mixed suppression pool air space and drywell after a period of 2 hours corresponding to the assumption of the in-vessel fuel melt source term. The licensee's assumption is consistent with the timing of the AST as described in RG 1.183 and, therefore, the NRC staff finds this approach acceptable.

3.1.1.1.3 Drywell Natural Deposition

Although the guidelines provided in RG 1.183, Appendix A, Section 3.2, allow for credit for the reduction in airborne radioactivity in the containment by natural deposition, the licensee did not take credit for natural deposition in the containment in the design-basis LOCA dose consequence analysis.

3.1.1.1.4 Drywell Spray Assumptions

RG 1.183, Appendix A, Section 3.3, states that, "Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited." In its LAR, the licensee based the credited spray removal on a spray pump volumetric flow rate of 5,600 gallons per minute (gpm). The licensee assumed that the spray would be initiated by manual action

20 minutes post-accident with an assumed termination at 4 hours and a fall height of 31 feet (ft) based on the difference in elevations between the lower spray head and the drywell floor.

The NRC staff examined the FitzPatrick UFSAR for evidence that the containment spray systems were designed to provide a reduction in airborne activity in accordance with SRP 6.5.2. Based on this examination, it appeared that the spray system was designed for pressure reduction and not specifically for reducing airborne radioactivity. The NRC staff noted that containment spray design requirements regarding the ability to reduce airborne radioactivity are discussed in Calculation No. JAF-CALC-19-00005, Revision 0, Section 12.3, NUREG-0800, Section 6.5.2 review items, which was submitted as part of the LAR. To address this concern, the NRC staff issued request for additional information (RAI) No. ARCB-RAI-1A requesting the licensee to provide additional information describing the design characteristics of the containment spray systems regarding the ability to provide a reduction in airborne activity in accordance with SRP 6.5.2, as discussed in Calculation No. JAF-CALC-19-00005, Revision 0, and how it will be incorporated into the FitzPatrick UFSAR.

In its letter dated March 30, 2020, the licensee responded to RAI No. ARCB-RAI-1A and explained that the FitzPatrick UFSAR will be updated in accordance with 10 CFR 50.71(e) as part of implementation of the approved amendment. The licensee provided a summary of the proposed changes as shown below:

- Sections 1.6.2.12, 4.8.4, 4.8.5, and 4.8.6.2 of the FitzPatrick UFSAR will be updated to include a discussion of how containment spray aids in removal of airborne fission products.

- Section 4.8.6.2 of the FitzPatrick UFSAR will be revised to summarize the design characteristics of the containment spray system that impact its ability to provide a reduction in airborne activity. The revision to Section 4.8.6.2 will include a discussion of how the requirements of ANS/ANSI 56.5 are met as they relate to calculation of airborne fission product removal following a LOCA such as geometry, physical features, flow characteristics, and mixing considerations as described in SRP Section 6.5.2.

The NRC staff examined the calculation of the particulate removal coefficient as documented in Calculation No. JAF-CALC-19-00005, Revision 0, on page 26. Based on this examination, it appeared that a spray drop fall height of 31 ft was determined by subtracting the elevation of the drywell floor from the elevation of the lower spray header. This method would be valid for the evaluation of particulate spray removal if there were no obstructions present in the drywell; however, this is not the case in a Mark I drywell. In addition, the analysis assumed a spray flow rate of 5,600 gpm. Assuming the full spray flow rate of 5,600 gpm would be valid for particulate spray removal if there were no obstructions present in the drywell; however, this is not the case in a Mark I drywell. The NRC staff noted that it is the unobstructed free-fall height that is of importance in the determination of the ability of the containment spray to effectively reduce airborne radioactivity. NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” Section H, discusses the issue of obstructions interfering with the effectiveness of sprays as follows:

H. Droplet-Structure Interactions

Reactor containment buildings are not simple, open volumes. Immediately below spray headers there is often a substantial open space. But, eventually, falling drops begin to encounter equipment, structures and

> operating floor of the reactor. The drywells of Mark I containments are well-known for the congestion that can interfere in the free fall of water droplets.
>
> The flooring in many reactor containments is grating or so-called "expanded sheet metal." Below the flooring are large volumes which, in a severe reactor accident, would hold aerosol-contaminated gas. It is of interest to know, then, if spray droplets, after hitting structures and the open flooring, would continue to sweep aerosols from the containment atmosphere. Certainly, in the case of the design basis analysis of iodine removal from containment atmospheres, it has been traditional to assume droplets are ineffective once they have hit a structure or the flooring.

To address these concerns, the NRC staff issued RAI ARCB-RAI-1B requesting that the licensee provide a justification for the use of the fall height of 31 ft and the full spray flow rate of 5,600 gpm in the determination of the particulate removal coefficient, which apparently did not consider obstructions present in the drywell.

In its letter dated March 30, 2020, the licensee explained that the LAR was based on Revision 0 of JAF-CALC-19-00005, which used a spray removal coefficient of 30.0 hour $(hr)^{-1}$ for times when the decontamination factor (DF) was ≤ 50. This value was based in part on a spray fall height calculated as the difference between the lower spray header elevation 287 ft (') - 6 inches (") and the bottom of drywell elevation (256' – 6"). The calculated value of 30.0 $hr^{-1}$ was then reduced by a factor of 10 to 3 $hr^{-1}$ when a DF of 50 was reached. The licensee stated that a reduction in the spray removal coefficient to account for possible obstructions to the spray coverage was not included in the evaluation.

The licensee referred to the methodology used by Nine Mile Point Nuclear Station, Unit 1 (Nine Mile Point 1 or NMP1) (ADAMS Accession No. ML070110231), and Oyster Creek Nuclear Generating Station (Oyster Creek) (ADAMS Accession No. ML050940234), which made specific reductions in the spray removal coefficient calculation based on obstructions in the drywell or blocked nozzles that may impede flow. The NRC staff's SE associated with the issuance of Amendment No. 194, dated December 19, 2007, for Nine Mile Point 1 regarding the implementation of alternative radiological source term (ADAMS Accession No. ML073230597), states the following:

> To account for "drywell congestion", the licensee [NMP1] multiplied the secondary spray flow rate by 0.67 for additional conservatism. Also, the fall height of 21.4 feet, used by the licensee [NMP Unit 1] conservatively reflects a one-third reduction to account for "drywell congestion."

In addition, the licensee referred to the three-dimensional modeling of the drywell performed for Oyster Creek, which resulted in a 33.3 percent reduction in fall height to account for obstructions and a 33.3 percent reduction in flow rate based on a modular accident analysis program analysis. Oyster Creek License Amendment Request No. 315 regarding application of AST, dated March 28, 2005 (ADAMS Accession No. ML050940234), stated, "Drywell congestion explicitly addressed by reduced spray flow and fall height."

The licensee stated that Nine Mile Point 1 used the same Oyster Creek assumption because it is expected that the obstructions would be similar for BWR Mark I containments. In its RAI response, the licensee provided a revised analysis of the spray fall height and spray flow rate

using the same methodology used by Nine Mile Point 1 and Oyster Creek. The revised analysis resulted in a reduction in the assumed particulate spray removal coefficients ($\lambda_{ps}$). The initial $\lambda_{ps}$ was reduced from the original value of 30 $hr^{-1}$ to 26.36 $hr^{-1}$ until the DF reaches 50, after which the value reduces by a factor 10 to 2.636 $hr^{-1}$. As a result of these input changes, there was a minimal increase in the LOCA doses calculated by the licensee. The NRC staff reviewed the changes made to the initial LAR regarding the adjustments made to account for the presence of obstructions within the drywell and concludes that the modifications are reasonable and, therefore, acceptable.

In its LAR, the licensee stated that:

> Based on the equations in Standard Review Plan 6.5.2, the elemental iodine removal coefficient is conservatively assumed to be the same as the particulate aerosol removal coefficient. When the elemental iodine inventory is reduced by a factor of 200 the spray removal of elemental iodine is terminated.

The NRC staff notes that SRP Section 6.5.2 limits the elemental iodine removal constant to a value of 20 $hr^{-1}$. The NRC staff performed confirmatory calculations using the RADTRAD program to assess the impact of including this limitation on the licensee's dose analysis and concluded that the impact was not significant for this case. Accordingly, the NRC staff finds this deviation from the SRP acceptable for this LAR. As discussed in the SRP, the elemental iodine removal constant should be limited to 20 $hr^{-1}$. Because the significance of this limitation to the dose calculation is case-specific, the NRC staff's finding in this SE is only applicable to this LAR.

3.1.1.1.5 Primary Containment Bypass

The FitzPatrick LOCA analysis assumes that all the containment leakage will leak directly to the environment bypassing the secondary containment during the RB drawdown period of 20 minutes. This activity leakage path is modeled as a ground level release from primary containment directly to the environment. After 20 minutes, the licensee assumes that all the containment leakage will mix with 50 percent of the RB volume, be filtered through the SGTS and exhausted to the plant stack at a flow rate of 6,600 standard cubic feet per minute (scfm). The allowable flow rates for the SGTS in TS 5.5.8, "Ventilation Filter Testing Program," are 5,400 to 6,600 (6,000±10%) scfm. Utilizing the maximum value is conservative as it maximizes the release. The licensee's assumption is consistent with the AST as described in RG 1.183 and, therefore, the NRC staff finds this approach acceptable.

3.1.1.1.6 Control of Suppression Pool Water Chemistry

The licensee took credit for pH control in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. The staff reviewed the licensee's analysis for maintaining suppression pool pH ≥ 7 for 30 days following a LOCA. According to RG 1.183, maintaining pH basic will minimize re-evolution of iodine from the suppression pool water.

According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide, which is a highly ionized salt soluble in water. Iodine in this form does not present any radiological problems since it remains dissolved in the sump water and

does not enter the containment atmosphere. However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can be, therefore, released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters of which pH is very important. Maintaining pH basic in the sump water will ensure that this conversion will be minimized. FitzPatrick proposes to control post-LOCA sump pH by injecting sodium pentaborate from the SLC system. The main purpose of the SLC system is to control reactivity in the case of control rod failure. However, sodium pentaborate can also act as a pH buffer. Such buffering action could maintain basic pH in the suppression pool despite the presence of strong acids.

In order to verify that the injection of sodium pentaborate from the SLC system will provide sufficient buffering action to maintain pH above 7, the licensee performed calculations using the methodology developed from NRC research reported in NUREG/CR-5950, “Iodine Evolution and pH Control.” Assumptions made by the licensee with respect to input parameters to the calculation were designed to produce the lowest possible calculated pH. By demonstrating that the suppression pool pH remains above 7 with all parameters conservatively adjusted to create the lowest possible pH condition, the licensee ensures that under realistic plant conditions, the pH threshold will not be challenged.

The licensee considered the effects of strong acid generation on the post-LOCA sump pH. Generation of strong acids inside containment is the primary phenomena that will lower suppression pool pH following a LOCA. Per guidance in NUREG/CR-5950, the licensee calculated the mass of nitric acid ($HNO_3$) generated by the radiolysis of air and water inside containment and the mass of hydrochloric acid generated by the radiolysis of Hypalon® electrical cable insulation inside containment. The staff verified that the licensee used conservative assumptions when calculating strong acid generation. For example, the licensee added well over 4,000 ft more cable material into the calculation than the amount that is actually present in the plant.

Minimum volume and concentration were used as calculation inputs for SLC system inventory. This, in conjunction with the conservative amounts of strong acids described above, resulted in a suppression pool pH of 8.12, 30 days after the postulated LOCA. The licensee also performed calculations simulating the post-LOCA pH without SLC system injection. In that scenario, it takes 72 hours for the pH to drop below 7. This demonstrates the amount of time available for operators to initiate SLC system injection to control pH. It should also be noted that the licensee provided calculations that demonstrate the SLC system inventory can be injected in less than 1 hour upon initiation.

The staff has independently verified the licensee’s calculations and finds that by using sodium pentaborate as a buffer, the pH of the suppression pool will remain above a pH of 7 for 30 days post-LOCA. The staff finds that following a LOCA, the re-evolution of iodine will be limited, given the buffering action of the sodium pentaborate injected by the SLC system to maintain the suppression pool pH above 7.0. Accordingly, the licensee proposed changes to TS 3.1.7, “Standby Liquid Control System,” and TS 3.3.6.1, “Primary Containment Isolation Instrumentation,” to expand the applicability of the TSs from Modes 1 and 2 to Modes 1, 2, and 3, and to expand the end state from Mode 3 to Mode 4 when the completion time for action statements is not met. Therefore, the NRC staff finds this approach acceptable.

3.1.1.1.7 Potential for Boron Precipitation and Degradation of Core Cooling

Implementation of the proposed AST LOCA radiological analysis at FitzPatrick will require use of the SLC system to control the pH level in the suppression pool during mitigation of AST LOCA. In BWRs, the SLC system is required to mitigate an anticipated transients without scram (ATWS) event in accordance with 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." The SLC system was designed as a backup method to maintain the reactor subcritical without control rods after an ATWS. Boric acid solution is stored in the SLC system tank and is injected directly into the lower plenum of the RPV as a means to shut down the reactor following an ATWS.

Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of 10 CFR does not require an SLC system, and it is not used to satisfy the 10 CFR 50.46 acceptance criteria during a design-basis LOCA. However, implementation of the AST LOCA radiological analysis at FitzPatrick will require use of the SLC system to control the pH level in the suppression pool during mitigation of AST LOCA. Because the SLC system will be relied upon for radiological analysis of AST LOCA at FitzPatrick, the objective of the staff's review is to examine whether use of boric acid solution from the SLC system following AST LOCA could result in boron precipitation and flow blockage in the core during the long-term cooling phase, causing degraded core cooling, to ensure the requirements of 10 CFR 50.46(b)(4) are met.

The licensee proposed to use the SLC system, which will inject sodium pentaborate solution into the lower plenum of the RPV where it will mix with the ECCS water and spill over to the drywell and then to the suppression pool. Sodium pentaborate is a base and will neutralize acids generated in the post-accident primary containment environment.

The NRC staff evaluated whether it is likely for boron injected from the SLC system to precipitate in the core, causing flow blockage and degraded core cooling. The staff determined that because the rates at which ECCS water injected by core spray (CS) at the top of the core and by residual heat removal pumps at the lower plenum of the vessel are substantially higher than the core boil-off rate, the boron solution is not expected to remain stagnant inside the core region as the boil-off occurs. Instead, the solution should flow out of the core and mix with the rest of the coolant. This should prevent boron concentration from rising significantly inside the core due to sustained boil-off. The colder water sprayed at the top of the core by CS should help keep the boron solution mixed inside the core by the natural circulation process. In addition, the fact that the boron solution remains very diluted and well-mixed throughout the period will make it unlikely that the boron concentration can rise to a level that can cause boron to precipitate inside the core any time during the long-term cooling phase.

Furthermore, the NRC staff noted that some of the findings made in a study performed by the Boiling Water Reactor Owners Group (BWROG), endorsed by the NRC staff and documented in the NRC letter dated June 29, 2018, "Closure of Potential Issues Related to Emergency Core Cooling Systems Strainer Performance at Boiling Water Reactors" (ADAMS Accession No. ML18078A061), are also applicable to potential issues related to boron precipitation if they occur during AST LOCA. The evaluation is documented in "BWROG Risk-Informed Debris Analysis – Staff Technical Evaluation," dated May 2018 (ADAMS Accession No. ML18058A602).

As part of the BWROG analysis, it was assumed that the fuel inlet filters become fully blocked with debris as soon as coolant reaches the fuel inlets during core reflood. The BWROG

performed thermal-hydraulic analyses using the General Electric-Hitachi Transient Reactor Analysis Code (TRACG code) (NEDE-33005P-A, "TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant Accident Analyses for BWR/2-6," dated February 24, 2017 (ADAMS Package Accession No. ML17055A387), to estimate realistic core temperatures (peak cladding temperatures) and determine the ECCS configurations required to provide cooling under various scenarios. The thermal-hydraulic analyses were used to determine whether core damage would occur for various conditions. The analyses found that for some scenarios, the low-pressure coolant injection pumps could provide adequate cooling. This determination depended on the number of pumps available and the size of the break. Notably, the analysis found that a single CS pump could provide adequate core cooling. Based on the BWROG evaluations, the NRC staff concluded that the effects of debris on fuel would not contribute significantly to increases in risk caused by the failure of long-term cooling for BWRs.

On the basis of the conclusions discussed above, the staff determined that in an unlikely scenario of boron precipitation and subsequent fuel inlet filter blockage by boron precipitates, one CS should provide adequate core cooling. Therefore, the staff concludes that boron precipitation and the resulting degradation of core cooling is not likely to occur during AST LOCA with SLC system injection at FitzPatrick. As such, the staff further concludes that the proposed approach is acceptable because it will continue to satisfy GDC 35 and 10 CFR 50.46(b)(4) insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any loss of coolant at a rate that fuel and clad damage that could interfere with continued effective core cooling will be prevented, and that the core remains amenable to cooling.

The staff concluded that SLC system injection during an AST LOCA will continue to meet the requirements of 10 CFR 50.46(b)(4) and GDC 35. The staff further concluded that adequate defense in depth will be maintained, and sufficient safety margins will be maintained. The staff, therefore, finds this approach acceptable.

3.1.1.2 Assumptions on Transport in Secondary Containment

The secondary containment structure (also referred to as the reactor building (RB)) completely encloses the primary containment structure such that a dual-containment design is utilized to limit the spread of radioactivity to the environment during a design-basis LOCA. The licensee has determined that following an accident initiation, the RB drawdown would be accomplished in 17.2 minutes. This value is determined by performing a GOTHIC calculation with conservative assumptions and inputs to maximize the drawdown time. For added conservatism in the dose consequence analysis, the licensee assumed an RB drawdown period of 20 minutes, at which time the secondary containment structure is maintained at a negative pressure, ensuring that leakage from primary containment to secondary containment can be collected and filtered by the SGTS prior to release to the environment. The SGTS performs the safety function of maintaining a negative pressure of 0.25 inches of vacuum water gauge within the secondary containment, as well as collecting and filtering the leakage from primary containment. In the LOCA analysis, the licensee credits the SGTS for mitigation of the radiological releases from the RB. After the assumed 20-minute drawdown, these releases are filtered by the SGTS and released by the SGTS exhaust vent. The LOCA analysis assumes an SGTS charcoal filter efficiency of 97 percent for all species of iodine. Accordingly, the licensee proposed to revise charcoal absorber efficiency in TS 5.5.8, "Ventilation Filter Testing Program (VFTP)," from 90 percent to 97 percent.

RG 1.183, Appendix A, Section 4.4, states that:

> Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%.

In accordance with RG 1.183, the licensee took credit for mixing within the RB prior to release. For conservatism, the licensee assumed an RB volume of 2,370,000 $ft^3$, which is less than the actual RB volume of 2,373,660 $ft^3$. The licensee then limited the RB mixing volume to 50 percent of the reduced RB volume, yielding an RB mixing volume of 1,185,000 $ft^3$. This approach is more conservative than the guidance in RG 1.183 and is acceptable to the NRC staff.

3.1.1.3 Assumptions on Engineered Safety Feature System Leakage

To evaluate the radiological consequences of engineered safety feature (ESF) leakage, the licensee used the deterministic approach as described in RG 1.183. This approach assumes that except for the noble gases, all the fission products released from the fuel mix instantaneously and homogeneously in the suppression pool water. Except for iodine, all the radioactive materials in the suppression pool are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of 5 percent released to the suppression pool water during the gap release phase and 35 percent released to the suppression pool water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the suppression pool concurrently. ECCS leakage develops when ESF systems circulate suppression pool water outside containment, and leaks develop through packing glands, pump shaft seals, and flanged connections.

The licensee determined that the maximum bulk suppression pool water temperature does not exceed 212 degrees Fahrenheit (°F). RG 1.183, Appendix A, Section 5.5, states that, “If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid.” Therefore, the licensee used a flash fraction of 10 percent to evaluate the contribution due to ESF leakage in the LOCA analysis. In accordance with RG 1.183, the licensee assumed that the chemical form of the released iodine is 97 percent elemental and 3 percent organic. All ESF leakage is assumed to be released into the secondary containment.

ESF leakage releases into the secondary containment during RB drawdown are assumed to leak directly to the environment as a ground level release with no filtration. After the assumed 20-minute drawdown period, ESF releases are assumed to be mixed in the secondary containment with a 50 percent mixing efficiency filtered by the SGTS and released by the SGTS vent. The licensee has taken no exception or departure from the guidance provided in RG 1.183 for evaluating the radiological consequence resulting from the ESF leakage fission product release pathway. Therefore, the NRC staff finds this approach acceptable.

3.1.1.4 Assumptions on Main Steam Isolation Valve Leakage

The MSLs in BWR plants, including FitzPatrick, contain dual quick-closing MSIVs. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, it is recognized that some leakage through the valves may occur. RG 1.183, Appendix A, Section 6, provides guidance for the evaluation of the radiological consequences from MSIV leakage, which should be combined with other fission product pathways to determine the total calculated radiological consequences from a LOCA.

Following the guidance in RG 1.183, the licensee assumed that the activity available for release from MSIV leakage is that activity determined to be in the drywell for evaluating containment leakage. The licensee did not credit activity reduction by the steam separators or by iodine partitioning in the reactor vessel. The wetwell free air volume was included with the drywell free air volume after 2 hours, as previously discussed in the containment leakage section of this SE.

In the FitzPatrick dose consequence analysis, total MSIV leakage is evaluated at a maximum leak rate of 270 standard cubic feet per hour (scfh) for all steam lines, and 135 scfh for any one line. These leakage rates are based on a pressure of 45 psig. The corresponding proposed TS SR 3.6.1.3.10 will limit combined MSL leakage rate to less than or equal to 200 scfh, and less than or equal to 100 scfh for any one line when tested at a pressure of 25 psig. The licensee credited a 50 percent reduction in the postulated maximum MSIV TS leakage after the first 24 hours based on a corresponding reduction in drywell pressure after 24 hours, as described in RG 1.183. The reduced MSIV leakage is assumed to continue for the duration of the accident evaluation period of 30 days.

The licensee converted MSIV leakage using the Ideal Gas Law to determine the actual leakage in cubic feet per hour (cfh) using the post-LOCA peak temperature and pressure. To account for the assumed mixing between the wetwell and drywell at 2 hours and the resulting activity dilution, the licensee reduced the flow rate through the MSIVs by the ratio of the drywell volume to the total volume at 2 hours. For added conservatism, the licensee did not credit this leak rate reduction in the determination of aerosol deposition in the MSLs.

RG 1.183, Section 5.1.2, “Credit for Engineered Safeguard Features,” states that:

> Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Following the guidance in RG 1.183 regarding the assumption of a single active component failure, the licensee assumed that the inboard MSIV in one of the shortest MSLs fails to close and remains open during the accident, which extends the well-mixed volume boundary from the RPV nozzle to the outboard MSIV.

RG 1.183, Appendix A, describes assumptions for evaluating the radiological consequences of a LOCA. Section 6 of Appendix A describes assumptions on MSIV leakage in BWRs. Assumption 6.5 states that:

> A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 [J.E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991. (ADAMS Accession Number ML003683718)] and A-10 [USNRC, "Safety Evaluation of GE Topical Report, NEDC-31858P (Proprietary GE report), Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993," letter dated March 3, 1999, ADAMS Accession Number 9903110303] provide guidance on acceptable models.

In its LAR, the licensee stated that all four MSL headers are Seismic Class I and Quality Assurance Category I from the RPV nozzle to the seismic boundary break at the turbine stop valve (TSV). Therefore, the licensee stated, and the NRC staff agrees, that since these lines are qualified to withstand the safe shutdown earthquake (SSE), they may be credited for aerosol deposition.

The licensee took credit for aerosol deposition in both the inboard and outboard section of the MSL between the outboard MSIV and the TSV. The licensee evaluated the total of 270 scfh MSIV leakage by assuming that the leakage would be equally divided between the two shortest MSLs. The licensee assigned an MSIV leakage of 135 scfh to the MSL with the failed open inboard MSIV by assuming one well-mixed volume between the RPV nozzle and outboard MSIV. The licensee assigned a second well-mixed volume in this line between the outboard MSIV and TSV.

The licensee credited aerosol deposition in the second shortest MSL between the RPV nozzle and inboard MSIV. The licensee modeled this section as one well-mixed volume between the RPV nozzle and inboard MSIV. The licensee modeled a second well-mixed volume between the inboard MSIV and TSV. The licensee limited credit for aerosol deposition to the horizontal sections of the MSLs, while the airborne elemental iodine in this release path is assumed to be adsorbed on the entire MSL surface area. The NRC staff finds that the licensee used modeling assumptions consistent with previous NRC-accepted practice for the evaluation of MSL aerosol deposition, and this aspect of its MSL aerosol deposition model is, therefore, acceptable.

<u>Assumption of a Main Steam Line Rupture</u>

In Attachment 1, Section 3.11.11 of the LAR, the licensee states that:

> The AST LOCA dose analysis implements a 20-group probabilistic settling velocity distribution for MSIV leakage rather than using the AEB 98-03 single, median value, model. The 20-group probabilistic distribution methodology has been previously approved at Clinton ([LAR] Reference 6.16), Limerick ([LAR] Reference 6.17), and LaSalle ([LAR] Reference 6.18). The same settling velocity

> probability distribution function shown in Equation 5 of AEB 98-03 is used to conservatively calculate aerosol settling velocity as follows…

The NRC staff notes that the cited precedents included a ruptured MSL to maximize the dose consequences from MSIV leakage. AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," included this assumption, as shown below:

> The staff's well-mixed deposition model assumes that each segment of piping in the RADTRAD nodalization is well-mixed. The unbroken main steam lines in the RADTRAD nodalization are modeled as two segments. The first segment is the length of piping between the reactor vessel and the first MSIV. The second segment is the length of piping between the first MSIV and the second MSIV. The broken main steam line is modeled as one segment of piping. This segment is the length of piping between the first MSIV and the second MSIV.

The licensee addressed this issue in Section 3.11.1, "Recirculation Line Rupture Versus Main Steam Line Rupture," of the LAR, which states that because the recirculation presents a greater challenge to selective aspects of facility design, a recirculation line rupture is the limiting event with respect to radiological consequences. The NRC staff notes that while it is true that mechanistically a recirculation line break would be expected to present a more significant challenge to the reactor core than a ruptured MSL, the source term used to satisfy 10 CFR 50.67 is a deterministic source term imposed on the facility to test the ability of systems to mitigate the releases sufficiently to meet predetermined acceptance criteria. Assuming a ruptured MSL in the evaluation of the acceptability of MSIV leakage criteria fulfills the underlying guidance from RG 1.183 that assumptions should be selected with the objective of maximizing the postulated radiological consequences.

The NRC staff notes that Calculation No. JAF-CALC-19-00005, Revision 0, Section 2.3.4, includes a discussion of the basis for assuming a recirculation line rupture instead of ruptured MSL in the assessment of MSIV leakage stating that:

> Although postulating a main steam line break in one steam line inside the drywell would maximize the dose contribution from the MSIV leakage, the steam line break is not a credible event during a LOCA, because the Seismic Class 1 main steam piping up to the TSVs is designed to withstand the SSE (Ref. 9.62).

The NRC staff notes that the integrity of the entire reactor coolant pressure boundary must comply with SSE requirements to satisfy Appendix A to 10 CFR Part 100. The assumption of a ruptured MSL for evaluating MSIV leakage, in conjunction with a deterministic source, does not imply a ruptured MSL in addition to a recirculation line rupture. Rather, the evaluation assumes a ruptured MSL (with a deterministic source term) to maximize the dose contribution from MSIV leakage.

To address this concern, the NRC staff issued RAI No. ARCB-RAI-3 requesting the licensee to justify that assuming a recirculation line rupture instead of an MSL rupture is consistent with the guidance from RG 1.183 that assumptions should be selected with the objective of maximizing the postulated radiological consequences.

In its letter dated March 30, 2020, the licensee responded by first stating that its analysis submitted in the LAR conservatively only modeled MSLs "B" and "C," which are symmetrical

and shorter than lines “A” and “D” and, therefore, would provide less volume for deposition. However, to address the NRC staff’s concern, the licensee provided the results of a separate analysis in which all four MSLs were modeled to quantify the dose consequences of assuming an MSL break in one of the lines. The licensee’s evaluation performed to model the MSL break also incorporated the changes in drywell spray removal discussed previously.

The licensee’s results demonstrated that the dose consequences based on modeling two steam lines and a recirculation line break as submitted in the LAR resulted in a slightly higher calculated dose for the CR than the analysis which modeled an MSL break in one line. The licensee stated that while the offsite doses for the ruptured MSL case were slightly higher than the analysis submitted in the LAR, the offsite doses remain well below the acceptance criteria. The NRC staff concludes that the analysis of the dose consequences resulting from an assumed MSL break provided by the licensee demonstrates that the impact of including an MSL break does not significantly impact the dose consequences from MSIV leakage for FitzPatrick.

Modeling of Aerosol Settling

The licensee stated that a long-established NRC staff concern is with the selection of a single value for MSL aerosol settling velocity to evaluate the deposition of aerosol particles having a wide range of particle sizes and weights. The degree of MSL deposition is expected to decrease as the leakage progresses through the lines, since the larger and heavier aerosols would have already settled out of the MSLs in upstream sections of piping. To address these concerns, the licensee employed the 20-group probabilistic distribution to assess the MSL aerosol settling velocity, which directly influences the aerosol deposition. The licensee stated that the 20-group probabilistic distribution methodology has been used in LARs that were subsequently approved for the Clinton Power Station, Unit 1 (Clinton or CPS); Limerick Generating Station, Unit 1 (Limerick or LGS); and LaSalle County Station, Units 1 and 2 (LaSalle). The following excerpts from the NRC staff’s SEs for those plants discuss the use of the 20-group probabilistic distribution method:

The SE associated with Amendment No. 167, dated September 19, 2005, for Clinton, Unit 1 (ADAMS Accession No. ML052570461), states that:

> AmerGen’s modeling of aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of an AST at the Perry Nuclear Power Plant. The aerosol settling model is described in a report, AEB-98-03, “Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term,” which was written by the NRC Office of Nuclear Regulatory Research. AEB-98-03 gives a distribution of aerosol settling velocities that are estimated to apply in the main steam line piping. The model used in the Perry assessment assumed aerosol settling may occur in the main steam lines upstream of the outboard MSIV, at the median settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. In the Perry assessment, aerosol settling is assumed to occur in one settling volume between the inboard MSIV and the outboard MSIV for the main steam line which has been assumed broken inside the drywell. For the remaining main steam lines aerosol settling is assumed to occur in two settling volumes; one between the reactor vessel and the inboard MSIV, the other volume between the two closed MSIVs.

AmerGen's modeling of aerosol settling in the MSIV leakage pathway for CPS is different from that for Perry in that piping downstream from the outboard MSIV to the auxiliary building/turbine building wall is credited, giving a larger volume and area of piping for settling. The NRC staff had requested information from the licensee regarding the conservatism in its use of the same settling velocity in the entire piping volume credited, including a third settling volume consisting of piping downstream of the outboard MSIV. The staff's concern was that the removal through aerosol settling was overestimated by modeling three settling volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping considering that the larger and heavier aerosols would have already settled out of main steam line atmosphere in upstream sections of piping. AmerGen responded by changing the model to assume only two settling volumes in an unbroken main steam line and one settling volume for the broken main steam line. The other two main steam lines are not assumed to leak, since the entire TS total allowable leakage of 200 scfh is assumed to occur through two lines at the maximum allowable rate of 100 scfh each. Additionally, AmerGen calculated a weighted average for the aerosol settling velocity from the AEB-98-03 distribution, converted that average settling velocity to an effective aerosol filtration efficiency for each main steam line, and applied the applicable effective filtration efficiency to the leakage rate out of each main steam line. The settling area was assumed to be the projected horizontal area in the horizontal sections of the qualified main steam piping. AmerGen did not take credit for aerosol settling after 24 hours, to address the change in the aerosol distribution over time.

Additional assumptions in AmerGen's modeling of the MSIV leakage pathway include; no credit for deposition beyond third MSIV, no credit for holdup or plate-out in the main condenser, one outboard MSIV fails to close on each line, 100 scfh leakage in each of the two shortest MSLs, the pressure between closed MSIVs is assumed to be equal to containment pressure and the temperature is assumed to be normal steam line conditions, and the pressure downstream of outboard MSIVs (and inboard MSIV on faulted line) is assumed to be atmospheric and steam is at normal operating temperature.

The staff acknowledges that aerosol settling is expected to occur in the main steam line piping but, because of recent concerns with aerosol sampling used in AEB-98-03 and lack of further information, does not know how much deposition (i.e., settling velocity value) is appropriate. The licensee [AmerGen] has used a model based on the methodology of AEB-98-03, but has applied additional conservatism to address the staff's comments on the applicability of the AEB-98-03 methodology to CPS as modeled. The staff has performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all pathways would continue to be acceptable even when the AEB-98-03 $10^{th}$ percentile settling velocity is assumed for the feedwater line aerosol settling. The $10^{th}$ percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the calculated settling velocities in AEB-98-03. Use of the $10^{th}$ percentile settling velocity is more conservative than use of the median settling velocity as noted as reasonable in AEB-98-03. Given this information, the staff finds the CPS main steam line aerosol settling model to be reasonable and appropriate.

The NRC staff notes that while the above-cited excerpt is silent on the 20-group method, this method was described in Clinton's RAI response dated August 19, 2005 (ADAMS Accession No. ML052430196), which states that:

> CPS is somewhat different than Perry in that piping downstream from the outboard MSIV to the Auxiliary Building/Turbine Building wall is credited. The NRC staff has questioned whether crediting the same settling velocities throughout the piping system, and treatment of the downstream piping as a third node was adequately conservative. In response, CPS:
>
> 1. Has combined penetration piping and downstream piping into a single outboard node;
>
> 2. Uses a 20 group probability distribution on settling velocities with settling efficiencies determined for each group and a net weighted average efficiency used, a process that is significantly more conservative than use of a median settling velocity;
>
> 3. Takes no credit [is taken] for aerosol settling after 24 hours.

The SE associated with Amendment No. 185, dated August 23, 2006, for Limerick, Units 1 and 2 (ADAMS Accession No. ML062210214), states that:

> Exelon's modeling of aerosol settling in the MSIV leakage pathway for LGS is different from that for Perry in that LGS credits five volumes (one pathway used two volumes in the steamline plus the condenser, the other pathway used one volume plus the condenser) for determining the settling deposition[1] while, in contrast, Perry credited three volumes.[2] The NRC staff questioned whether assuming the same settling velocities throughout the piping system and the condenser was adequately conservative. The staff's concern was that the removal through aerosol settling was overestimated by modeling all volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping. This lesser settling effect is due to larger and heavier aerosols that settle out of the main steamline atmosphere in upstream sections of piping.
>
> Exelon responded by changing its model to combine penetration piping and downstream piping into a single outboard node. Additionally, Exelon used a 20-group probability distribution of settling velocities with efficiencies determined for each group and a net weighted average efficiency (a process that Exelon states is significantly more conservative than use of a median settling velocity). Exelon did not take credit for aerosol settling after 24 hours, to address the change in the aerosol distribution over time.
>
> The NRC staff acknowledges that aerosol settling is expected to occur in the main steamline piping, however, because of recent concerns regarding the AEB-98-03 report and the lack of further information, does not know how much deposition (i.e., which settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, but included some additional conservatism to attempt to address the NRC staff's questions on the applicability of the AEB-98-03 methodology to LGS. The NRC staff has

performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all main steamline pathways would continue to be acceptable, even when the AEB-98-03 10th-percentile settling velocity is assumed for bypass aerosol settling. The 10th-percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the calculated settling velocities in AEB-98-03. Based upon AEB-98-03, use of the 10th-percentile settling velocity is more conservative than use of the median settling velocity noted as reasonable in AEB-98-03. Given this information, and the presence of a seismically-qualified condenser, the NRC staff finds the LGS main steamline aerosol settling model to be reasonable.

[1] This is treated as an effective filtration efficiency in both models.
[2] No credit is taken for deposition in the MSIV alternate drain pathway to the condenser.

The SE associated with Amendment No. 197, dated September 6, 2010, for LaSalle, Units 1 and 2 (ADAMS Accession No. ML101750625), states that:

The licensee's modeling of aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of an AST at the Perry Nuclear Power Plant (Perry). The aerosol settling model is described in a report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," which was written by the NRC Office of Nuclear Regulatory Research. AEB-98-03 gives a distribution of aerosol settling velocities that are estimated to apply in the main steamline piping. The model used in the Perry assessment assumed aerosol settling may occur in the main steamlines at the median settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. In the Perry assessment, aerosol settling is assumed to occur in one settling volume downstream of the outboard MSIV for one main steamline. For the remaining modeled line, settling is assumed to occur in two settling volumes; one between the two closed MSIVs and one downstream of the outboard MSIV.

The licensee's modeling of aerosol settling in the MSIV leakage pathway for LSCS [LaSalle County Station] is different from that for Perry in that piping downstream from the outboard MSIV to and including the condenser is credited. Additionally, the licensee used a 20 group probability distribution of settling velocities with efficiencies determined for each group and a net weighted average efficiency (a process that the licensee states is significantly more conservative than use of a median settling velocity). The licensee did not take credit for aerosol settling after 24 hours, to address the change in the aerosol distribution over time.

The NRC staff acknowledges that aerosol settling is expected to occur in the main steamline piping, but because of recent concerns regarding the AEB-98-03 report and the lack of further information, the staff does not know how much deposition (i.e., which settling velocity value) is appropriate. The licensee used a model based on the methodology in AEB-98-03, but included some additional conservatism to address the NRC staff's concerns about the applicability of the AEB-98-03 methodology to LSCS. The 10th percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the

> calculated settling velocities in AEB-98-03. Based upon AEB-98-03, use of the 10th percentile settling velocity is more conservative than the use of the median settling velocity noted as reasonable in AEB-98-03. Given the aforementioned conservatism and the presence of a seismically qualified condenser, the NRC staff finds the LSCS main steamline aerosol settling model to be reasonable and appropriate.

In the FitzPatrick LAR, the licensee stated that the modeling of aerosol settling is based on the methodology used by the NRC staff in its review of the implementation of an AST at the Perry Nuclear Power Plant (Perry). However, because of concerns regarding the aerosol settling model described in the report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," the licensee included some additional conservatism to address the NRC staff's concerns about the applicability of the AEB-98-03 methodology to FitzPatrick.

AEB-98-03 gives a distribution of aerosol settling velocities estimated to apply in the MSL piping. In the Perry assessment, aerosol settling is assumed to occur in one settling volume between the inboard MSIV and the outboard MSIV for the MSL, which has been assumed to be broken inside the drywell. For the remaining MSLs aerosol, settling is assumed to occur in two settling volumes – one between the reactor vessel and the inboard MSIV and the other volume between the two closed MSIVs.

The licensee's modeling of aerosol settling in the MSIV leakage pathway for FitzPatrick is different from that for Perry. At FitzPatrick, all four MSL piping sections between the RPV nozzle and the TSVs used in the MSIV leakage release paths remain intact and are capable of performing their safety function during and following an SSE, and thereby comply with RG 1.183, Appendix A, Section 6.5 guidance. The licensee stated that since the MSL piping up to the TSVs is designed to withstand an SSE, the horizontal pipe surface area and the volume up to the TSVs may be credited in the aerosol removal calculation. In addition, the licensee applied the 20-group method to compute probability distributions of settling velocities and removal efficiencies for each group rather than using the AEB 98-03 single median value model.

The licensee asserted that the 20-group methodology simulates the varied population of aerosol particulates having uneven settling velocities between the heavier and larger particles versus the lighter and smaller particles in a given MSL volume. The purpose is to more realistically account for the uneven settling of "easier to remove particles" versus "difficult to remove particles" in the consequence analysis. It models the distribution of aerosol particles and subsequence settling velocities as a semi-continuous, probability-weighted 20-group step function for each MSL volume. As the aerosol particles move from one volume to another, the distribution of particles is recalculated and compared to the initial distribution. This process successively shifts "weight" from the easier to remove particles when entering the piping to the difficult to remove particles as flow moves through the MSL. When the aerosol particulates finally exit the system to the environment, the recalculated probability distribution indicates a much more likely chance of "seeing" difficult to remove particles than was the case when entering the system.

The 20-group methodology utilizes Equation 5 of AEB 98-03 to develop the aerosol settling velocity distribution for the MSIV leakage pathway. As applied by the licensee, the settling velocity distribution is a function of this equation evaluated over a randomly sampled range of the three critical aerosol parameters (i.e., density/weight (logarithmically distributed), diameter/size (uniformly distributed), and shape (uniformly distributed)), and three constants

(i.e., gravitational acceleration, Cunningham slip factor, and viscosity). As typical for these calculations, the settling velocity distribution was generated using 10,000 randomly generated histories. Each of the 10,000 calculated settling velocities was given a probability of 1/10,000th, thereby making the cumulative fraction total equal to 1. The 20-group step function is then developed to approximate the continuous settling velocity distribution function calculated from the 10,000 histories.

The licensee computed the aerosol particulate release fractions for each of the 20 groups for each volume by applying Equations 2 and 3 of AEB 98-03. These equations account for the settling velocity, settling area, volumetric flow rate, and the volume of the well-mixed region being modeled. From each volume, the particulate release fraction is subtracted by 1 to convert it into "removal efficiencies." The set of 20 removal efficiencies calculated for each volume is then combined to form a set of 20 "net release fractions" for a given MSL. The net release fractions associated with a given group are the product of the removal efficiencies for each volume and the probability associated with that specific settling velocity group. Finally, the set of 20 net release fractions is summed and again subtracted from 1 to calculate a total effective aerosol removal efficiency, or TEARE, for input to the RADTRAD code. This process is performed for each MSL being modeled by recalculating each distribution exiting a volume and then using the result to calculate the distribution entering the next volume.

In Attachment 1, Section 3.11.11 of the LAR, the licensee states:

> The AST LOCA dose analysis implements a 20-group probabilistic settling velocity distribution for MSIV leakage rather than using the AEB 98-03 single, median value, model. The 20-group probabilistic distribution methodology has been previously approved at Clinton (Reference 6.16), Limerick (Reference 6.17), and LaSalle (Reference 6.18). The same settling velocity probability distribution function shown in Equation 5 of AEB 98-03 is used to conservatively calculate aerosol settling velocity as follows….

The NRC staff notes that the analyses cited as precedents did not credit drywell sprays. Page 96 of NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," provides details on how sprays impact aerosols. NUREG/CR-5966 indicates that the sprays shift the sizes of aerosols in the containment towards those that are removed most slowly (the mean aerosol size decreases as the sprays operate). The licensee's estimates of aerosol deposition in the steam lines is determined using, in part Equation 5 of AEB 98-03. Equation 5 provides the aerosol settling (and thus the aerosol deposition) in the steam line and indicates that the aerosol settling is proportional to the square of the diameter of the aerosols. Because the sprays shift the size of the aerosols to smaller sizes, the aerosols settling in the steam lines would decrease due to these smaller diameter aerosols.

In the Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2 or NMP2) LAR (ADAMS Accession No. ML071580314) seeking to incorporate 10 CFR 50.67 into the Nine Mile Point 2 licensing basis, Calculation H21C-106, Revision 0, page C1 discusses a "penalty" on the sedimentation velocity (aerosol settling velocity) used for bypass pathways to account for the recognition that the sprays (preferentially) remove large particles in primary containment. Also, information previously provided by Nine Mile Point 2 discussed the impact of containment sprays on aerosol deposition in the steam line (see the NRC staff's RAI Question 7 and the Nine Mile Point 2 response to this RAI in a letter dated January 7, 2008 (ADAMS Accession No. ML20149K682), stating that the settling velocity was chosen "to reflect the substantial spray removal credited in the Nine Mile Point 2, containment."

As discussed in the SE for Amendment No. 215, dated May 29, 2008 for Nine Mile Point 2 (ADAMS Accession No. ML081230439), the NRC staff stated that it had issues with the use of AEB 98-03 for modeling aerosol deposition for Nine Mile Point 2. In that SE, the staff stated that the licensee used a settling velocity of 0.000066 m/sec to address the staff's issues regarding the use of AEB 98-03 and that this value was sufficiently conservative (along with other conservatisms) to reflect the effectiveness of the sprays.

The issue of how the change in the aerosol size due to drywell sprays would impact assumptions made in the subsequent MSL aerosol deposition was discussed in the FitzPatrick pre-application meeting held on June 20, 2019 (ADAMS Accession No. ML19183A128). As the staff stated in the meeting summary, "Since the precedent cited for the proposed MSL aerosol deposition did not include drywell sprays, the licensee should consider including a detailed discussion of how the use of sprays is accounted for in the subsequent steam line aerosol deposition."

The NRC staff acknowledges that aerosol settling is expected to occur in the main steamline piping. The NRC staff further acknowledges that the 20-group method was used in previous LARs that were found to be acceptable by the staff. However, since these previous evaluations did not credit aerosol removal from drywell sprays and the licensee did not provide a discussion of the impact of drywell sprays on the subsequent MSL deposition, the NRC staff concluded that additional information was needed regarding the licensee's MSIV leakage assessment. Therefore, as discussed below, the staff requested additional information regarding the impact of drywell sprays on MSL deposition.

3.1.1.4.1 Aerosol Removal by Sprays and in Main Steam Lines

From an examination of the submitted information in the LAR, it appears that the licensee considers the aerosol removal by sprays and aerosol removal in the MSLs as independent removal mechanisms. The NRC staff notes that regardless of the specific removal mechanisms involved, larger aerosol particles in the containment atmosphere will be preferentially removed, making subsequent removal by deposition in downstream piping more challenging. To address this concern, the NRC staff issued RAI No. ARCB-RAI-2 requesting the licensee to provide additional information describing how the gravitational settling credited in the MSLs considers the changing aerosol characteristics (i.e., aerosol size and density distributions) due to the preferential removal of larger aerosols because of the credit assigned to containment sprays.

In its letter dated March 30, 2020, the licensee responded by submitting the results of a sensitivity analysis, which was intended to evaluate the impact of sprays on the aerosol settling velocity and the impacts on the analysis of including a ruptured steam line. In its response, the licensee identified other inputs that could be used to offset the limitations associated with the current aerosol deposition model in the LAR. The licensee stated that its sensitivity analysis concluded that other identified considerations are sufficient to offset the uncertainty introduced by the drywell spray effects on the aerosol deposition model.

3.1.1.4.2 Licensee's Sensitivity Analysis

In its letter dated March 30, 2020, the licensee stated that there are other significant conservatisms associated with the AST LOCA model that were not included in its sensitivity analysis. For instance, the licensee stated that CR atmospheric dispersion factors have readily defined uncertainty distributions that if incorporated would demonstrate there is a substantial

amount of margin in the associated input parameters. The NRC staff notes, however, that the use of conservative atmospheric dispersion factors in design-basis dose consequence analyses is a well-established practice. Atmospheric dispersion factors are based on the evaluation of site-specific meteorological data. These data are processed to provide values at the 95 percent confidence level, ensuring that there is reasonable assurance that the acceptance criteria will not be exceeded. Therefore, the staff does not endorse the concept of including reduced values for atmospheric dispersion factors in dose consequence sensitivity analyses. The NRC staff, therefore, agrees with the licensee's exclusion of such factors in its sensitivity analyses.

The licensee stated that among the many conservatisms used in dose consequence analyses, the following conservatisms were deemed to be reasonable to define and model deterministically:

- credit full drywell spray lambdas (not included in the licensee's evaluation)
- credit for plateout and deposition in drywell (not included the licensee's evaluation)
- inclusion of all four MSLs for holdup and deposition
- more realistic CR operator breathing rate
- aerosol impaction on the first closed MSIV
- condenser holdup and deposition

The licensee developed a model using insights from NUREG/CR-5966 to quantify the impact of drywell sprays on the assumed aerosol settling velocity and subsequently on aerosol deposition in the MSLs. The licensee's evaluation acknowledged that sprays will remove aerosols at different rates depending on their particle size. The licensee quantified the suspended aerosol mass for 20 distinct particle size groups based on a probability distribution. The licensee did not credit the interaction of particles with one another that could result in an agglomeration resulting in larger particle sizes and subsequently higher deposition rates. Each particle size group was evaluated independently.

The licensee assumed a 2-micron aerosol mass median diameter and geometric standard deviation of 2.0 particle size in the 20-group method to recalculate the aerosol removal rates in its sensitivity analysis. The licensee claimed that this modeling assumption is conservative and results in a much smaller gravitational settling and spray removal rate. The NRC staff did not review and evaluate this assumption because: (1) no basis was provided for the assumption, (2) this assumption was not used in the licensee's proposed analysis of record, and (3) it was not used by the NRC staff to determine reasonable assurance for complying with 10 CFR 50.67 for this particular LAR.

The licensee's sensitivity analysis model is intended to account for the effect of the drywell spray reducing the size of the particles, resulting in smaller gravitational settling velocities and spray removal rates than in the model submitted in the LAR. Based on this analysis, the licensee recalculated the dose consequences and presented the results as a "base sensitivity case." The licensee then adjusted the base sensitivity case to account for more realistic CR operator breathing rates, aerosol impaction on the first closed MSIV, and holdup and deposition in the condenser. The following discussion describes the NRC staff's assessment of each of the elements of the licensee's sensitivity analysis.

3.1.1.4.3 NRC Staff's Assessment of the Elements Used in the Licensee's Sensitivity Analysis

3.1.1.4.3.1 Control Room Operator Breathing Rate

In its letter dated March 30, 2020, the licensee referenced breathing rate data from Table 6-17 of Environmental Protection Agency (EPA) handbook EPA/600/R-09/052F, "Exposure Factors Handbook: 2011 Edition." Table 6-17 provides breathing rates as a function of age for various percentiles up to a maximum value. RG 1.183 recommends using a constant value of $3.4 \times 10^{-4}$ $m^3$/sec for the duration of the CR dose consequence analysis. For its sensitivity analysis, the licensee used the RG 1.183 recommended for the first 2 hours, followed by a reduced value from the EPA handbook of $3.28 \times 10^{-4}$ $m^3$/sec from 2 to 12 hours and $3.06 \times 10^{-4}$ $m^3$/sec from 12 hours to 30 days. The licensee chose the 95 percent confidence level values from the EPA Handbook for light intensity work typical of a CR operator. As a result of these modifications, the CR dose was reduced slightly (approximately a 5 percent reduction). The NRC staff notes that while the use of a breathing rate for light intensity work might be justified during time periods of normal working conditions, it should not be considered under 10 CFR 50.67 for determining radiation exposures from "access to and occupancy" of the CR under accident conditions, when CR personnel may be expected to be at a higher level of stress and engaged in increased activities. Therefore, the NRC staff disagrees with the licensee's selection of a reduced breathing rate in the sensitivity analysis. Nonetheless, this factor has only a small effect, as the licensee's sensitivity analysis shows a small effect of the breathing rate on the dose consequence (5 percent reduction).

3.1.1.4.3.2 Aerosol Main Steam Isolation Valve Impaction

In its letter dated March 30, 2020, the licensee referenced the Nine Mile Point 1 AST LOCA licensing basis described in Nine Mile Point 1 AST Calculation H21C092 (ADAMS Accession No. ML070110240), which credits the phenomenon of impaction at the first closed MSIV. The licensee explained that in this scenario, some of the aerosol particles will be deposited on the MSIV sealing surface as the aerosols entrained with the carrier gas leak through the closed MSIV. Nine Mile Point 1 conservatively determined this impaction results in a DF of 2, which is modeled as a 50 percent filter in the transfer pathway through the first closed MSIV. This reduction is only accounted for once in each MSL. The licensee referred to previously approved Amendment No. 125 for Nine Mile Point 1, dated May 29, 2008; this reference; however, is for Unit 2 and does not support the licensee's justification. The NRC staff acknowledges, however, that Nine Mile Point 1 included the following assumption in its MSIV leakage dose consequence analysis:

> Assumption 7 from, Nine Mile Point Unit 1 Alternative Source Term Calculation H21C092 "UI LOCA w/LOOP, AST Methodology" (ADAMS Accession No. ML070110240) [Non-proprietary]: "It is assumed that aerosol reaching the first closed valve in RB bypass pathways (including MSIV leakage) experiences a DF of 2 due to impaction. This applies also to the adsorbed elemental iodine.

The NRC staff notes that the NRC approval cited in the licensee's RAI response of March 30, 2020, was for Nine Mile Point 1. The SE associated with Amendment No. 194, dated December 19, 2007, for Nine Mile Point 1, states that:

> The NRC staff believes that, though there is merit to this plugging phenomenon and impaction in theory, there is not enough empirical evidence, directly related to the unique and hypothetical conditions associated with a design-basis LOCA

> event, to warrant full credit for such a considerable DF attributable to impaction. Therefore, the NRC staff does not generally endorse taking credit for impaction when modeling removal of particulates in main steam lines following a LOCA. However, the NRC staff does believe that enough evidence exists to verify the conservatism of a DF of 2 in the specific design-basis LOCA model at NMP1. The contribution of this impaction DF to the overall iodine activity decontamination, does not lead to an excessive overall credit for iodine removal in the MSLs. Based on the approximate DF of 4 that credits for removal by sedimentation (See Section 3.2.1.2.1.4), combined with this DF of 2, the licensee is assuming less than a 90% overall iodine removal efficiency in the steam lines. If this MSIV leakage pathway were modeled using a well-mixed model, as described and previously approved in AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," December 9, 1998, the calculated activity removal in the MSLs would be analogous to that calculated by the licensee. Therefore, the NRC staff finds the overall iodine removal credited by to be acceptable, as modeled for NMP1.

The cited excerpt from the Nine Mile Point 1 2007 SE thus clearly states that the NRC staff does not generally endorse taking credit for impaction when modeling removal of particulates in MSLs following a LOCA. Although the SE concluded that the overall iodine removal credited as modeled for Nine Mile Point 1 was acceptable, this conclusion was based upon specific aspects of the Nine Mile Point analysis, which was very conservative, and should not be interpreted as NRC staff acceptance of credit for impaction when modeling removal of particulates in MSLs following a LOCA. Consistent with the views expressed by the staff in its SER regarding the Nine Mile Point AST analysis, the NRC staff does not consider credit for MSIV impaction to be appropriate for use in the FitzPatrick MSIV leakage sensitivity analysis submitted in the licensee's letter dated March 30, 2020.

3.1.1.4.3.3 Condenser Holdup and Deposition

In its letter dated March 30, 2020, the licensee stated that aerosol holdup and deposition provided by the condenser is not modeled in JAF-CALC-19-00005 for FitzPatrick and that depending on the event scenario, multiple pathways could exist to route activity to the condenser, including the drain lines and the turbine itself.

In its sensitivity analysis, the licensee modeled an MSIV leakage pathway to the condenser through the drain lines from the MSL piping between the MSIVs. The licensee stated that this model did not credit any holdup and deposition in the outboard MSL piping and that modeling the release to the condenser from the piping between the MSIV is consistent with other plants in the Exelon fleet (e.g., LaSalle and Limerick). The licensee further stated that operating experience associated with the North Anna Power Station earthquake and post-Fukushima evaluations have shown that components and piping systems typically used in this release path are sufficiently rugged to ensure they are capable of performing some level of radioactivity removal during and following an SSE. Therefore, the licensee concluded that it is reasonable to assume that the condenser pathway could be made available for mitigating the consequences of MSIV leakage.

3.1.1.4.4 NRC Staff Evaluation of Licensee's Sensitivity Analysis

In response to the NRC staff's RAI, the licensee performed a total of eight sensitivity cases for FitzPatrick by adjusting the base case discussed previously, which uses more conservative MSL deposition assumptions to account for the influence of drywell sprays. The sensitivity cases employed various combinations of the use of the EPA value for breathing rate, credit for MSIV impaction, assessment of the impacts of a ruptured steam line, and consideration of the holdup and deposition of fission products in the condenser. The licensee's base case indicates that the conservative modeling of the drywell spray on the aerosol removal in the MSLs without making other adjustments results in increased doses. The NRC staff notes that the base case results indicate that while the calculated CR dose exceeds the 5 rem acceptance criterion, the offsite doses to members of the public remain a small fraction (less than 10 percent) of the acceptance criteria. The NRC staff notes that the licensee's base case was produced for the purpose of conducting a sensitivity analysis and is not intended to replace the accident analysis of record, which is the revised analysis discussed in the letter dated March 30, 2020, in response to the RAI concerning obstructions (ARCB-RAI-1B). The analysis of record indicates that dose consequences comply with all applicable dose acceptance criteria.

The licensee stated that in order to analyze the effect of drywell sprays, simplifications of the aerosol physics were made in their modeling that resulted in calculated aerosol removal constants that are substantially lower, when compared to values typically produced using more accurate assessments. The NRC staff compared the licensee's aerosol removal constants used in the base case with the values used in other MSIV leakage analyses submitted by other licensees for NRC staff review and agrees that the values used in the licensee's base case are conservative.

The licensee's sensitivity results demonstrate that the condenser is very effective in substantially reducing the dose consequences from MSIV leakage. While other elements assessed in the licensee's sensitivity analysis provided relatively small decreases in the calculated doses, the licensee's analysis of the condenser's mitigation properties demonstrated a substantial dose reduction. The NRC staff notes that the guidance provided in RG 1.183 for design-basis LOCA radiological analysis states that structures, systems, and components (SSCs) may be credited with creating a pathway to the condenser if they are able to withstand an SSE. However, the staff also considers it reasonable to include the probability of the existence of a pathway to the condenser to offset uncertainties in crediting aerosol removal from drywell sprays in calculating the dose consequences of MSIV leakage. The NRC staff's consideration of risk and engineering insights is discussed in Section 3.5 of this SE.

In addition, the probability of an event resulting in substantial fuel melt such as a LOCA followed by a significant unrelated seismic event is very low. Specifically, RG 1.183, Appendix A, describes assumptions for evaluating the radiological consequences of a LOCA. Section 6 of that appendix describes acceptable assumptions on MSIV leakage in BWRs. Assumption 6.5 states:

> A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 [J.E. Cline, "MSIV Leakage Iodine Transport Analysis,"

Letter Report dated March 26, 1991. (ADAMS Accession Number ML003683718)] and A-10 [USNRC, "Safety Evaluation of GE Topical Report, NEDC-31858P (Proprietary GE report), Revision 2, BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993," letter dated March 3, 1999, ADAMS Accession Number 9903110303.] provide guidance on acceptable models.

The dose acceptance for the LOCA analysis described in RG 1.183 is based on the regulatory acceptance criteria for what is commonly termed the maximum hypothetical accident (MHA) or the maximum credible accident. The MHA is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite dose consequences evaluated based on design-basis containment leakage.

The NRC staff notes that, by design, an SSE would not result in any core damage. The design-basis radiological assessment of an MHA (referred to in guidance as a LOCA) deterministically imposes a fuel melt source term into the containment to test the ability of the plant to meet predetermined dose acceptance criteria. Since the SSE would not cause fuel damage, the exclusion of non-SSE qualified SSCs in the dose analysis implies that two independent extraordinary events could occur during the analysis period: an event resulting in substantial fuel melt followed by a significant unrelated seismic event. The probability of this sequence of events is very low.

3.1.1.5 Control Room Habitability

Under accident conditions, habitability for the CR is provided by the control room emergency ventilation air supply system (CREVAS). This system provides habitability zone isolation and a positive pressure for the CR. The licensee credits CREVAS actuation at 30 minutes post-accident. 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 licensee included the following contributions to CR dose from the following DBA-LOCA radiation sources:

- contamination of the CR atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility

- contamination of the CR atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the CR envelope

- radiation shine from the external radioactive plume released from the facility

- radiation shine from radioactive material in buildings adjacent to the control structure, which includes the containment, the RB, and the turbine building (TB)

- radiation shine from radioactive material in systems and components inside or external to the CR envelope (e.g., CR filter shine)

The first two elements of CR dose listed above were evaluated by the licensee with the RADTRAD program, which evaluates these contributions directly. In order to assess the direct shine dose elements listed, the licensee used the RADTRAD program for the determination of the unattenuated dose and then applied attenuation factors to account for the significant concrete shielding (30 inches thick) of the walls and ceiling of the CR.

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% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be $3.5 \times 10^{-4}$ cubic meters per second.

The NRC staff reviewed the licensee's assessment and concludes the licensee appropriately considered all relevant contributions to CR dose from the postulated LOCA dose.

### 3.2 Atmospheric Dispersion Estimates

Onsite CR and offsite exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion values ($\chi$/Qs) for FitzPatrick were previously approved by the NRC in the SE for Amendment No. 261, dated April 14, 2000, for changes to the TSs regarding the allowed containment leakage rate (ADAMS Accession No. ML18031A041).

Regulatory Position 5.3 of RG 1.183 states that "Atmospheric dispersion values ($\chi$/Qs) for the EAB, the LPZ, and the CR that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide." In accordance with this guidance, the licensee stated that the atmospheric dispersion values ($\chi$/Qs) for the EAB, the LPZ, and the CR that were previously approved by the staff for the previously analyzed release pathways are used in the LOCA analysis.

In addition to those previously approved $\chi$/Qs, in the LAR, the licensee proposed new DBA $\chi$/Qs for the new ground level TB's MSIV leakage release pathway to the CR and Technical Support Center (TSC). The determination of the new CR and TSC $\chi$/Q values were made using the ARCON96 atmospheric dispersion model (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," ADAMS Accession No. ML17213A187), pursuant to the guidance of RG 1.194. The NRC staff reviewed the licensee's new atmospheric dispersion analyses as described below.

### 3.2.1 Meteorological Data

The licensee provided supplemental information in its letter dated August 27, 2019, 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 were formatted for the ARCON96 atmospheric dispersion code in order to calculate updated χ/Q values for the CR and TSC. This format contained hourly data on wind speed, wind direction, and atmospheric stability class.

The staff previously completed a detailed review related to the acceptability and representativeness of the 1985 through 1992 onsite hourly meteorological data for FitzPatrick Amendment No. 261 dated April 14, 2000, and Amendment No. 276, dated September 12, 2002, regarding TS changes to the requirements for handling irradiated fuel assemblies (ADAMS Accession No. 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.

### 3.2.2 Onsite Control Room and Technical Support Center Atmospheric Dispersion Estimates

In support of the LAR, the licensee used the computer code ARCON96 to estimate χ/Q values for the CR and TSC for potential accidental releases of radioactive material. RG 1.194 endorses the ARCON96 model for determining χ/Q values to be used in the design-basis evaluations of CR radiological habitability.

The ARCON96 code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days. The meteorological input to ARCON96 consists of hourly values of wind speed, wind direction, and atmospheric stability class. The χ/Q values calculated through ARCON96 are based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the release points and receptors. The diffusion coefficients account for enhanced dispersion under low-wind speed conditions and in building wakes.

The hourly meteorological data are used to calculate hourly relative concentrations. The hourly relative concentrations are then combined to estimate concentrations ranging in duration from 2 hours to 30 days. Cumulative frequency distributions prepared from the average relative concentrations and the relative concentrations that are exceeded no more than 5 percent of the time for each averaging period are determined.

The dispersion coefficients used in ARCON96 have three components. The first component is the diffusion coefficient used in other NRC models such as PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," ADAMS Accession No. ML12045A149). The other two components are corrections to account for enhanced dispersion under low-wind speed conditions and in building wakes. These components are based on analysis of diffusion data collected in various building wake diffusion experiments under a wide range of meteorological conditions. Because the dispersion occurs at short distances within the plant's building complex, the ARCON96 dispersion parameters are not affected by nearby topographic features such as bodies of water. Therefore, the staff finds the licensee's use of the ARCON96 dispersion parameter assumptions acceptable for use in estimating χ/Q values for the CR and TSC for potential accidental releases.

Attachment 9 of the LAR dated August 8, 2019, includes the inputs and assumptions used for the ARCON96 calculations of the new χ/Q values. The attachment includes the meteorological file names, the lower and upper height measurements, and units of wind speed. Attachment 9 also contains a description of the input values of vertical velocity, stack flow, stack radius, and diffusion coefficients. The release type, release height, and building area for each release pathway are also listed. The attachment also includes the distances to receptor, intake height, elevation difference, and direction to source for each release pathway. Finally, Table 2 in Section 8 of Attachment 9 lists the new CR and TSC χ/Q values calculated with ARCON96 for the 0-2, 2-8, 8-24, 24-96, and 96-720-hour time intervals.

The staff confirmed the licensee's CR and TSC atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. Both the staff and licensee used the same release assumption for each of the release pathway-receptor combinations, as well as the previously discussed source-receptor distances, directions, heights, and area values. The staff also confirmed that the analysis and assumptions were consistent with the guidance of RG 1.194. Based on the results of its confirmatory analysis and the fact that the licensee followed NRC guidance, the staff finds the licensee's CR and TSC χ/Q values acceptable for use in the radiological consequence assessments.

#### 3.2.3 Atmospheric Dispersion Conclusions

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 pathways. The staff found that the licensee used methods consistent with regulatory guidance identified in Section 2.0 of this SE. The licensee used onsite meteorological data that complied with the guidance of RG 1.23. The inputs and assumptions used to calculate the CR and TSC 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 consequences assessments associated with this LAR.

### 3.3 Environmental Qualification of Electric Equipment

#### 3.3.1 Alternative Source Term and Environmental Qualification

The licensee proposed adopting the AST in accordance with 10 CFR 50.67 for use in calculating the LOCA dose consequences. Section 6 of RG 1.183 states that the NRC staff at that time was assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has since been resolved, as documented in a memo dated April 30, 2001 (ADAMS Accession No. ML011210348), and in Supplement 25 to NUREG-0933, Generic Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification," June 2001 (ADAMS Accession No. ML012190402).

As stated in the conclusion to Generic Issue 187:

> The staff concluded that there was no clear basis for backfitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would

potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary. Thus, the issue was DROPPED from further pursuit.

Therefore, in consideration of the cited references, the NRC staff finds that it is acceptable for the TID-14844 AST to remain the licensing basis for EQ.

The licensee used the methodology contained in TID-14844 to determine the radiation doses in the existing EQ analyses for FitzPatrick. The use of this methodology is consistent with the guidance contained in RG 1.183. The staff confirmed that the licensee will continue to use the TID-14844 methodology as a result of the LAR. Based on this, the staff finds that no new electrical equipment needs to be added to the licensee's 10 CFR 50.49 EQ program and that the EQ of electrical equipment should remain bounded due to full-scope implementation of the AST.

3.3.2 Main Steam Isolation Valve Leakage Rate and EQ

The proposed LAR would increase the MSIV leakage rate in TS SR 3.6.1.3.10. Previously, the MSIV leakage pathway was not considered because the MSLC system would direct any MSIV leakage to the SGTS. The new release pathway for the MSIV leakage is the TB. The staff evaluated whether electrical equipment in the TB would remain bounded by the existing EQ due to the proposed change.

The licensee provided an evaluation of the radiological impact on the EQ of electrical equipment due to the proposed increased leakage rate of the MSIVs. However, the licensee did not provide an evaluation of the impact of the MSIV increased leakage rate on temperature, pressure, or humidity of electrical equipment in the TB. Therefore, the staff requested the licensee to provide additional information that showed that the temperatures, pressures, and humidity remain bounded by the existing EQ for electrical equipment in the TB that are impacted by the proposed change. In its letter dated January 16, 2020, the licensee stated that it performed an evaluation to demonstrate that the current EQ design conditions bound the increased mass and energy that would enter the TB due to not crediting the MSLC system to process MSIV leakage. The licensee's evaluation showed that the smallest high-energy line break (HELB) over the design period of 10 minutes results in mass and energy releases more than two orders of magnitude greater than that of the MSIV leakage during the 8-hour release before the TB ventilation system is restarted. Therefore, the temperatures, pressures, and humidity remain bounded by the existing EQ for electrical equipment and components in the TB that are impacted by the proposed change. Based on its review of the licensee's response, the staff finds that the licensee's evaluation adequately shows that the temperature, pressure, and humidity remain bounded by the existing EQ for electrical equipment in the TB as a result of the proposed change.

Upon reviewing the LAR, it was unclear to the staff as to whether the licensee considered the impact of the proposed change on non-safety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. Therefore, the staff requested additional information on how the licensee assessed the impact of the proposed change on non-safety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. In its letter dated January 16, 2020, the

licensee stated that with the exception of the electric bays and cable tunnels, the TB is currently considered a harsh environment for temperature, pressure, and humidity during an HELB. Thus, any non-safety-related electrical components whose failure could prevent satisfactory accomplishment of a safety function must be scoped into the EQ program for the postulated HELB. According to the licensee, there are currently no EQ components located in the TB due to either a safety-related function during a HELB or a failure with potential to prevent accomplishment of a safety-related function. Based on its review of the licensee's response, the staff determined there is reasonable assurance that the proposed change will not adversely affect the potential for non-safety-related electrical equipment to prevent satisfactory accomplishment of safety functions.

The staff further requested the licensee to confirm whether any components are being added to the EQ equipment list to comply with 10 CFR 50.49 due to the proposed change. In its letter dated January 16, 2020, the licensee stated that there are no components that are being added to the EQ equipment list to comply with 10 CFR 50.49 due to the proposed change. Based on the licensee's response, the staff determined that the licensee will not be adding any new electrical equipment to the licensee's 10 CFR 50.49 EQ program as a result of the proposed change to the MSIV leakage rate. Hence, the licensee's EQ equipment list remains the same and continues to comply with 10 CFR 50.49. The staff finds this acceptable.

The staff finds that the EQ of electrical equipment will remain bounded upon full-scope implementation of the AST. Based on its review of the information in the LAR, as well as the additional information provided by the licensee, the staff finds that the EQ of electrical equipment will not be adversely impacted by the proposed changes. Therefore, the staff finds the proposed changes acceptable.

## 3.4 Technical Specification Changes

### 3.4.1 Main Steam Isolation Valve Leakage Rate Limits

The proposed revision to SR 3.6.1.3.10 would increase the combined total leakage rate for all four MSIV leakage paths from ≤ 46 scfh to ≤ 200 scfh when tested at ≥ 25 psig, with a limitation that leakage through any one MSIV steam line cannot exceed 100 scfh.

The test pressure of ≥ 25 psig in the TSs is less than the peak calculated accident pressure, $P_a$, of 45 psig in TS 5.5.6, "Primary Containment Isolation Valves (PCIVs)." The licensee used a total MSIV leakage rate of 270 scfh for the four lines in the AST analysis. As noted in Attachment 1, Table 3.6-1, "General AST Parameter or Method," of the LAR, leakage at 200 scfh when tested at 25 psig is equivalent to 270 scfh at 45 psig. The maximum allowed limit for MSIV leakage in the proposed TSs is consistent with the MSIV leakage limits used in the reanalysis of the postulated LOCA radiological consequences, with adjustment as necessary for test pressure. Since the proposed change to the TS MSIV leakage rate limits are derived from the reanalysis of the postulated LOCA radiological consequences, the NRC staff finds there is reasonable assurance that (1) 10 CFR 50.36(b) will continue to be met, and (2) 10 CFR 50.36(c)(3) will continue to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs are met. Therefore, the proposed revision to SR 3.6.1.3.10 is acceptable.

The regulation at 10 CFR 50.36(a)(1) states, in part, "A summary statement of the bases or reasons for such specifications … shall also be included in the application but shall not become part of the technical specifications." Accordingly, along with the proposed TS changes, the

licensee also submitted TS Bases changes that correspond to the proposed TS changes for information only. The licensee will make supporting changes to the TS Bases in accordance with TS 5.5.11, "Technical Specifications (TS) Bases Control Program."

3.4.2 TS 3.6.1.8, "Main Steam Leakage Collection (MSLC) System"

The licensee proposes to delete TS 3.6.1.8 in its entirety. TS 3.6.1.8 LCO requires two MSLC subsystems to be operable in Modes 1, 2, and 3; has associated action statements that address conditions when the LCO is not met; and has two SRs that assure the necessary quality of the MSLC system and components is maintained, that facility operation will be within safety limits, and that the LCO will be met.

As described in the FitzPatrick UFSAR, Section 9.19, "Main Steam Leakage Collection System," the safety objective of the MSLC system is to collect and direct leakage past the MSIVs following a LOCA for processing by the SGTS so that resultant exposures are maintained below the radiological dose values specified by 10 CFR Part 100. In the LAR, the licensee stated that a LOCA reanalysis was performed, which implements an AST methodology in accordance with 10 CFR 50.67, and that upon NRC approval, the current design-basis source term assumptions and radiological consequences for the LOCA will be replaced with the new AST LOCA analysis. In the AST LOCA analysis, the MSLC system is no longer credited for the mitigation of any DBA to maintain the resultant radiological dose below the values in 10 CFR 50.67.

The regulation at 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the four criteria listed in Section 2.4.5 of this SE.

The MSLC system is designed to collect and process leakage across the seats of the MSIVs and to collect and process stem packing leakage from the outboard containment MSIVs following a design-basis LOCA. Leakage across the seat and from the stem packing is considered normal valve leakage and, therefore, the MSLC system does not provide any detection of abnormal degradation of the reactor coolant pressure boundary. The MSLC, therefore, does not meet 10 CFR 50.36(c)(2)(ii) Criterion 1.

The AST LOCA analysis assumes the total MSIV leakage for all four MSLs is 270 scfh at 45 psig, MSIV leakage through any one steam line is 135 scfh at 45 psig, and the release is to the environment as an unfiltered ground level release by the TB. The MSIV leakage pathways are unfiltered ground level releases, and the MSLC system is not used to collect and process these pathways. Therefore, the MSLC system does not provide a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The MSLC, therefore, does not meet 10 CFR 50.36(c)(2)(ii) Criterion 2.

Currently, the MSLC system would be actuated following a DBA to mitigate the consequences of the accident by directing any MSIV leakage to the SGTS. However, the AST LOCA analysis demonstrates that the offsite and onsite radiological dose consequences following a DBA are acceptable and meet the requirements of 10 CFR 50.67 without operation of this system. Therefore, the MSLC system is no longer credited as part of the primary success path and no longer mitigates a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The MSLC, therefore, does not meet 10 CFR 50.36(c)(2)(ii) Criterion 3 for the AST.

The discussion of Criterion 4 in the Final Policy Statement on TSs improvements for nuclear power reactors states that it is the Commission's policy that licensees retain in their TS LCOs action statements and SRs for the following systems, which operating experience and probabilistic risk assessment has generally shown to be important to public health and safety:

- reactor core isolation cooling/isolation condenser
- residual heat removal
- standby liquid control
- recirculation pump trip

The MSLC system is not listed in the Final Policy Statement as important to public health and safety, and neither recent operating experience nor probabilistic risk assessment has shown this system to be risk significant. Additionally, the inoperability of this system has no impact on core damage frequency or large early release frequency. The AST LOCA analysis demonstrates that the offsite and onsite radiological dose consequences of the DBA are acceptable and meet the requirements of 10 CFR 50.67 without operation of the MSLC system. Consequently, the MSLC system is not significant to the public health and safety and, therefore, does not meet 10 CFR 50.36(c)(2)(ii) Criterion 4.

Because the MSLC system no longer meets the criteria in 10 CFR 50.36(c)(2)(ii) to establish a TS LCO, the NRC staff finds it acceptable to delete TS 3.6.1.8 LCO and its associated action statements upon adoption of the AST.

The regulation at 10 CFR 50.36(c) states, in part: "Technical specifications will include items in the following categories." Paragraph 50.36(c)(3) of 10 CFR states:

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

Since TS 3.6.1.8 LCO no longer meets the requirements in 10 CFR 50.36 and is deleted from the FitzPatrick TSs, the NRC staff finds it acceptable to delete its associated SRs, as they no longer support a TS LCO.

3.4.2.1 Conclusion Regarding TS 3.6.1.8, "Main Steam Leakage Collection (MSLC) System"

The NRC staff reviewed the proposed deletion of TS 3.6.1.8 and concludes that this TS no longer meets the requirements of 10 CFR 50.36(a), 10 CFR 50.36(c)(2)(i), 10 CFR 50.36(c)(2)(ii), and 10 CFR 50.36(c)(3) for the reasons discussed above; thus, the NRC staff concludes that deletion of TS 3.6.1.8 in its entirety is acceptable. As a result of this change, the licensee proposes to update the TS Table of Contents page to reflect the deletion of TS 3.6.1.8. This change is editorial in nature and is, therefore, acceptable.

3.4.3 <u>TS 3.1.7, "Standby Liquid Control (SLC) System," and Associated Instrumentation in TS 3.3.6.1, "Primary Containment Isolation Instrumentation"</u>

The licensee proposes to add Mode 3 to the applicability statement in TS 3.1.7, add Mode 3 to the applicable modes or other specified conditions column in TS Table 3.3.6.1-1 for its associated instrumentation, and add TS 3.1.7 Required Action C.2 to be in Mode 4 instead of

stopping at Mode 3 if the required actions and associated completion time for Conditions A or B are not met.

The regulation at 10 CFR 50.36(b) requires that each license authorizing reactor operation include TSs derived from the analyses and evaluation included in the safety analysis report and amendments thereto. Currently, the SLC system is not credited for post-LOCA pH control. The new AST LOCA analysis assumes credit for control of pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. Therefore, the SLC system is required to maintain the suppression pool pH above 7 so that re-evolution of iodine from the suppression pool does not occur post-LOCA. SR 3.1.7.1 and Figure 3.1.7.1 in TS 3.1.7 ensure that this function is maintained in the current applicable modes. However, LOCAs are possible in Modes 1, 2, and 3 when there is considerable energy in the reactor core. In Modes 4 and 5, the probability and consequence of this event is reduced due to the pressure and temperature limitations of these modes.

Because LOCAs are possible in Modes 1, 2, and 3, the licensee proposes to add Mode 3 to the applicability statement in TS 3.1.7 and to the applicable modes or other specified conditions column in TS Table 3.3.6.1-1 for its associated instrumentation – Function 5.d, “SLC System Initiation.” The NRC staff finds the addition of Mode 3 to the applicability statement in TS 3.1.7 and to the applicable modes or other specified conditions column in TS Table 3.3.6.1-1 for Function 5.d to be a more restrictive change because it requires the SLC system and its instrumentation to be operable in an additional mode and requires SR performance in Mode 3. The NRC staff also finds that there is considerable energy in the reactor core in Mode 3, that LOCAs are possible, and the addition of Mode 3 is consistent with the new AST LOCA analysis. Therefore, the NRC staff concludes the addition of Mode 3 to the applicability statement in TS 3.1.7 and to the applicable modes or other specified conditions column in TS Table 3.3.6.1-1 for Function 5.d is acceptable, and the requirements in 10 CFR 50.36(b) will continue to be met.

Currently, TS 3.1.7 Condition C requires within 12 hours that the plant be placed in Mode 3 when the required action and associated completion time of Condition A or B is not met. The licensee proposes to add TS 3.1.7 Required Action C.2 which states, “Be in MODE 4,” with a completion time of 36 hours. If the SLC system cannot be restored to operable status within the required completion time for Condition A or B, the plant must be brought to a mode in which the LCO does not apply. To achieve this status, the plant must be brought to at least Mode 4. In Mode 4, the probability and consequence of LOCAs is reduced due to the pressure and temperature limitations of this mode. The NRC staff finds that the addition of TS 3.1.7 Required Action C.2 to place the unit in Mode 4 is reasonable because the probability and consequence of LOCAs is reduced due to the pressure and temperature limitations in Mode 4. The 36-hour completion time is reasonable based on operating experience to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Based on the above, the NRC staff concludes that the addition of TS 3.1.7 Required Action C.2 to be in Mode 4 within 36 hours provides acceptable remedial actions as allowed by 10 CFR 50.36(c)(2) to ensure that the SLC system is capable of performing its safety functions, and is, therefore, acceptable.

In accordance with 10 CFR 50.36(a)(1), the licensee submitted TS Bases changes that correspond to the proposed TS changes for information only. The licensee will make supporting changes to the TS Bases in accordance with TS 5.5.11, “Technical Specifications (TS) Bases Control Program.”

### 3.4.4 TS 5.5.8, "Ventilation Filter Testing Program (VFTP)"

The licensee proposes to revise the VFTP testing requirements in TS 5.5.8 for the SGTS and the CREVAS HEPA filters and the charcoal absorbers. The TS 5.5.8.a requirement for the in-place test of the HEPA filters to show a penetration and system bypass less than 1.0 percent is being revised to less than 1.5 percent. The requirement in TS 5.5.8.c that a laboratory test of a sample of the charcoal absorber demonstrate a methyl iodide penetration less than 5 percent is being revised to 1.5 percent when tested in series. In Section 7.7 of Attachment 6 to the LAR, the licensee stated that the SGTS and CREVAS HEPA and charcoal absorber filter efficiencies were calculated to be 97 percent in accordance with the guidance in Generic Letter 99-02; the NRC staff independently confirmed that to be the case. The proposed changes to the testing requirements ensure that the desired performance of the HEPA and charcoal absorber is consistent with the inputs in the revised radiological consequence analysis. The NRC staff also confirmed that the flow rates used for the SGTS and CREVAS, as indicated in Table 3.11-1 of Attachment 1 to the LAR, are consistent with the flow rates in the existing VFTP.

The regulation at 10 CFR 50.36(c)(5) requires TSs to include items in the category of administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. TS 5.5.8 is included in the Administrative Controls section of the licensee's technical specifications.

The licensee's calculation of the SGTS charcoal and HEPA filtration efficiencies for use as inputs in the dose consequence analysis is in Attachment 6, Section 7.7 of the LAR. The licensee states that these proposed changes are necessary to make the testing requirements consistent with the revised design-basis AST LOCA analysis.

Since the proposed changes to TS 5.5.8 VFTP are derived from the reanalysis of the AST LOCA radiological consequences, the NRC staff finds there is reasonable assurance that (1) 10 CFR 50.36(b) will continue to be met, and (2) 10 CFR 50.36(c)(5) will continue to ensure operation of the facility in a safe manner. Therefore, the proposed changes to TS 5.5.8 are acceptable.

### 3.4.5 TS 3.6.4.1, "Secondary Containment"

SR 3.6.4.1.1 requires the secondary containment vacuum to be ≥ 0.25 inch of vacuum water gauge when the secondary containment is required to be operable. A note is being added to SR 3.6.4.1.1. The note allows the SR to not be met for up to 4 hours if an analysis demonstrates that one standby gas treatment subsystem can establish the required secondary containment vacuum. During normal operation, conditions may occur that result in SR 3.6.4.1.1 not being met for short durations. For example, wind gusts that lower external pressure or loss of the normal ventilation system that maintain secondary containment vacuum may affect secondary containment vacuum. These conditions may not be indicative of degradations of the secondary containment boundary or of the ability of the SGTS to perform its specified safety function.

The note provides an allowance for the licensee to confirm secondary containment operability by confirming that one SGT subsystem can perform its specified safety function. This confirmation is necessary to apply the exception to meeting the SR acceptance criterion. While the duration of these occurrences is anticipated to be very brief, the allowance is permitted for a

maximum of 4 hours, which is consistent with the time permitted for secondary containment to be inoperable per Condition A of LCO 3.6.4.1.

The proposed note is consistent with TSTF-551, Revision 3, “Revise Secondary Containment Surveillance Requirements.” The licensee stated that in the current licensing basis radiological dose consequence analysis, the secondary containment pressure is assumed to be below atmospheric pressure at the time the LOCA event occurs. The analysis also assumed that the SGTS automatically starts and maintains negative pressure in secondary containment such that no exfiltration occurs during the event. Based on this consideration, FitzPatrick was previously approved to change SR 3.6.4.1.3 to state “verify one secondary access door in each access opening is closed, except when the access opening is being used for entry and exit.” Regarding SR 3.6.4.1.1, the licensee stated that TSTF-551 did not apply to FitzPatrick as a result of its existing dose analysis assumptions.

In the LAR, the licensee stated that a plant-specific analysis was performed to determine the secondary containment drawdown time using GOTHIC 8.2 to calculate secondary containment temperature and pressure response. As stated in Attachment 1, Section 3.12, “Applicability of TSTF-551 Safety Evaluation,” to the LAR, the calculation was based on RG 1.183, Appendix A, Section 4.3 assumptions regarding outside summer and winter conditions and wind speeds. Four different cases were analyzed with the GOTHIC model: summer with no wind, summer with wind, winter with no wind, and winter with wind. The sequence of events considered a DBA LOCA to occur coincident with loss-of-offsite power (LOOP), diesel generators starting and only one SGTS operating after a single failure of the second SGTS, SGTS flow at minimum value of 5,600 scfm as in TS 5.5.8, and a time of 21 seconds for the SGTS valves to fully open. The results indicated that it would take 17.2 minutes for the differential pressure inside secondary containment with respect to outside to reach - 0.25 inches water gage.

The proposed addition of the note to SR 3.6.4.1.1 does not change the TS requirement to meet SR 3.6.4.1.4. 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 ≤ 6,000 cubic feet per minute. In addition, TS LCO 3.6.4.3, “Standby Gas Treatment (SGT) System,” must be met; otherwise, the licensee shall shut down the reactor or follow any remedial action permitted by TSs until the condition can be met.

As discussed above, secondary containment operability is based on its ability to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. To prevent ground level exfiltration of radioactive material, the secondary containment pressure must be maintained at a pressure that is less than atmospheric pressure. The secondary containment requires support systems to maintain the control volume pressure less than atmospheric pressure. Following an accident, the SGTS ensures the secondary containment pressure is less than the external atmospheric pressure. During normal operation, non-safety-related systems are used to maintain the secondary containment at a negative pressure. However, during normal operation, it is possible for the secondary containment vacuum to be momentarily less than the required vacuum for several reasons. These conditions may not be indicative of degradations of the secondary containment boundary or of the ability of the SGTS to perform its specified safety function. Since the licensee meets the requirements of SR 3.6.4.1.4, meets the LCO or is following the actions of TS LCO 3.6.4.3, and the licensee’s analysis confirms secondary containment operability by confirming that one SGT subsystem can perform its specified safety function, there is reasonable assurance that the secondary containment and SGT subsystem will maintain the vacuum requirements during a DBA.

The NRC staff has determined that the conditions that may cause momentary reductions in secondary containment vacuum do not affect (1) the ability to maintain the secondary containment pressure during an accident at a vacuum that is consistent with the accident analyses, and (2) the time assumed in the accident analyses to drawdown the secondary containment pressure; the staff, therefore, concludes that the secondary containment can perform its safety function and may be considered TS operable. This is evident by being able to successfully perform and meet SR 3.6.4.1.4. These SRs require the SGTS to establish and maintain the required vacuum in the secondary containment as assumed in the accident analyses.

Furthermore, because the specified safety functions of the secondary containment and SGT subsystem can be performed in the time assumed in the licensee's accident analysis, then the fission products that bypass or leak from primary containment or are released from the reactor coolant pressure boundary components located in secondary containment prior to release to the environment will be contained and processed as assumed in the licensee's design-basis radiological consequence dose analyses. Based on the revised secondary containment drawdown time of 17.2 minutes determined by the GOTHIC analysis, the revised radiological consequence analysis used 20 minutes of unfiltered ground level release from secondary containment. The NRC staff finds that the proposed change is acceptable.

The NRC staff clarifies that the addition of the note to SR 3.6.4.1.1 is not required as a result of the revised dose consequence analysis. Rather, the analysis afforded the licensee the ability to add the note and benefit from the flexibility it offered in addressing the SR 3.6.4.1.1 being not met for short durations that are not indicative of degradation of the secondary containment boundary.

In accordance with 10 CFR 50.36(a)(1), the licensee submitted TS Bases changes that correspond to the proposed TS changes for information only. The licensee will make supporting changes to the TS Bases in accordance with TS 5.5.11, "Technical Specifications (TS) Bases Control Program."

### 3.4.6 Summary of Technical Specification Changes

The NRC staff reviewed the proposed TSs changes and determined that they meet the requirements for TSs in 10 CFR 50.36(b) because they are derived from the analyses and evaluation included in the safety analysis report and the amendment thereto. Additionally, the changes to the TSs were reviewed for technical clarity and consistency with customary terminology and format in accordance with SRP Chapter 16.0. The NRC staff reviewed the proposed TSs changes against the regulations and concludes that they meet the requirements of 10 CFR 50.36(a)(1), 10 CFR 50.36(c)(2)(i), 10 CFR 50.36(c)(2)(ii), 10 CFR 50.36(c)(3), and 10 CFR 50.36(c)(5) for the reasons discussed above, and thus, provide reasonable assurance that the TSs will have the requisite requirements and controls for the plant to operate safely. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

## 3.5 NRC Staff Risk and Engineering Insights

The LAR was not submitted as a formal "risk-informed" submittal with probabilistic risk assessment information in accordance with the guidance of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the

Licensing Basis" (ADAMS Accession No. ML17317A256). Thus, the NRC staff's findings are primarily based on traditional deterministic review approaches.

In the SRM to SECY-19-0036, the Commission directed the staff to apply risk-informed principles in any licensing review or other regulatory decision when strict, prescriptive application of deterministic criteria is unnecessary to provide for reasonable assurance of adequate protection of public health and safety. Risk-informed principles are consistent with the Commission direction in the SRM for SECY-19-0036. Since the application is not a fully risk-informed submittal (with probabilistic risk information), the staff does not apply risk as the basis for acceptance of a request; however, the following risk and engineering insights inform the technical review by supporting the deterministic safety conclusions and enhance the technical reviewers' confidence in their technical evaluations.

As described in Section 3.1.1.4.3.3, the licensee stated that aerosol holdup and deposition provided by the condenser is not modeled in JAF-CALC-19-00005 and that depending on the event scenario, multiple pathways could exist to route activity to the condenser, including the drain lines and the turbine itself. The licensee concluded that it is reasonable to assume that the condenser pathway could be made available for mitigating the consequences of MSIV leakage.

The NRC staff performed an independent assessment evaluating the capability of the power conversion system (PCS) and main condenser to serve as a holdup volume for MSIV leakage. The staff evaluated the seismic capacity of the SSCs in the PCS, including the main steam piping, equalization header, and main condenser, to assess whether they would be available to provide a holdup volume for fission products following an SSE. The staff used engineering information such as operations and design knowledge, as well as probabilistic and risk information to complete the evaluation. The staff also leveraged more recent relevant operating experience such as that obtained from the March 11, 2011, magnitude 9.0 Great East Japan Earthquake that caused the accident at the Fukushima Dai-ichi Nuclear Power Plant and the August 23, 2011, magnitude 5.8 earthquake near the North Anna Power Station. The staff's independent assessment found it is reasonable to conclude that the SSCs in the PCS would be available following an SSE and that the likelihood of them being unavailable to serve as a volume for holdup and retention is very low.

The assessment provides an insight when addressing uncertainties in the calculation of the dose consequences of MSIV leakage. Specifically, the staff recognizes that there is a high probability that doses will be significantly lower than those estimated using deterministic methods that do not credit holdup and retention of the MSIV leakage within the PCS.

Based on the available information and assessments using conservatively biased assumptions about the seismic capacity of the SSCs in the realistic pathway, the staff determined that there is high confidence that the MSLs and the PCS will be available for fission product dilution, holdup, and retention, especially at the seismic accelerations at a plant's design-basis SSE. Conservatisms and risk insights result in additional safety margin. In addition, as mentioned in the statements of consideration for 10 CFR 50.67, defense in depth is addressed using a DBA in the deterministic dose calculation. Therefore, consistent with the statements of consideration for 10 CFR 50.67, the principles of risk-informed decisionmaking, and the Commission direction to the staff in the SRM to SECY-19-0036, the NRC staff has determined these risk and engineering insights support its reasonable assurance finding based on its deterministic review.

### 3.6 NRC Staff Conclusion on Alternative Source Term

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of a LOCA representing a full implementation of an AST at FitzPatrick. The NRC staff finds that the licensee used analysis, methods, and assumptions that meet the intent of the regulatory requirements and guidance identified in Section 2.0 above. The NRC staff concludes there is reasonable assurance supported by risk and engineering insights, that the licensee's estimates for the EAB, LPZ, and CR doses comply with the cited acceptance criteria. The NRC staff further finds reasonable assurance that FitzPatrick, as modified by this license amendment, will continue to operate with sufficient safety margins and adequate defense in depth to address unanticipated events. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of a LOCA.

This licensing action is considered a full implementation of the AST. With this approval, the previous AST in the FitzPatrick design-basis LOCA dose consequence analysis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria described in 10 CFR 50.67. All future LOCA dose consequence radiological accident analyses that are performed to show compliance with regulatory requirements should address all characteristics of the AST and employ TEDE acceptance criteria as defined in the FitzPatrick design basis and modified by the present amendment.

## 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 April 14, 2020. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 (November 19, 2019; 84 FR 63899). 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.

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Appendix:
Table 1: FitzPatrick Loss-of-Coolant Accident (LOCA) Radiological Consequences Expressed as TEDE [(1)](rem)
Table 2: FitzPatrick Control Room (CR) Atmospheric Dispersion Factors
Table 3: FitzPatrick Offsite Atmospheric Dispersion Factors (sec/m3)
Table 4: FitzPatrick Control Room (CR) Data and Assumptions
Table 5: FitzPatrick Data and Assumptions for the Loss-of-Coolant Accident (LOCA)

# APPENDIX

**Table 1**
**FitzPatrick Loss-of-Coolant Accident (LOCA) Radiological Consequences**
**Expressed as TEDE [1] (rem)**

| LOCA Release Pathways | LOCA TEDE Dose in rem Receptor Location | | |
|---|---|---|---|
| | CR | EAB [2] | LPZ [3] |
| Containment Leakage | 1.06 | 0.51 @ 3.2 hours | 0.3 |
| ESF Leakage | $9.68 \times 10^{-2}$ | $9.51 \times 10^{-2}$ @ 9.8 hours | $3.49 \times 10^{-2}$ |
| MSIV Leakage | 3.44 | 0.22 @ 8.4 hours | 0.27 |
| Containment Shine to CR | $9.56 \times 10^{-3}$ | N/A | N/A |
| External Cloud to CR | 0.065 | N/A | N/A |
| CR Filter Shine | Negligible | N/A | N/A |
| Total Dose | 4.67 | 0.83 | 0.60 |
| Acceptance Criteria | 5 | 25 | 25 |

[1] Total effective dose equivalent
[2] Exclusion area boundary maximum 2-hour dose
[3] Low population zone 30-day dose at the outer boundary

**Table 2**
**FitzPatrick Control Room (CR) Atmospheric Dispersion Factors**

| Source Location/Duration | | χ/Q (sec/m$^3$) |
|---|---|---|
| For evaluating releases from containment leakage and ESF leakage via SGTS | | |
| | 0 - 8 hours | 9.26 x10$^{-7}$ |
| | 8 - 24 hours | 6.75 x10$^{-7}$ |
| | 24 - 96 hours | 3.39 x10$^{-7}$ |
| | 96 - 720 hours | 1.26 x10$^{-7}$ |
| For evaluating releases from containment leakage and ESF leakage before RB drawdown | | |
| | 0 - 2 hours | 3.52 x10$^{-3}$ |
| | 2 - 8 hours | 3.31 x10$^{-3}$ |
| | 8 - 24 hours | 1.43 x10$^{-3}$ |
| | 24 - 96 hours | 7.73 x10$^{-4}$ |
| | 96 - 720 hours | 6.07 x10$^{-4}$ |
| For evaluating MSIV leakage (turbine building (TB) release) | | |
| | 0 - 2 hours | 4.52 x10$^{-3}$ |
| | 2 - 8 hours | 3.33 x10$^{-3}$ |
| | 8 - 24 hours | 1.19 x10$^{-3}$ |
| | 24 - 96 hours | 8.27 x10$^{-4}$ |
| | 96 - 720 hours | 6.40 x10$^{-4}$ |
| For evaluating MSIV leakage for External Cloud Dose (TB release)[1] | | |
| | 0 - 2 hours | 4.52 x10$^{-3}$ |
| | 2 - 8 hours | 3.33 x10$^{-3}$ |
| | 8 - 24 hours | 1.19 x10$^{-3}$ |
| | 24 - 96 hours | 8.27 x10$^{-4}$ x 0.6 = 4.96 x10$^{-4}$ |
| | 96 - 720 hours | 6.40 x10$^{-4}$ x 0.4 = 2.56 x10$^{-4}$ |
| CR operator breathing rate (0 – 720 hours) hours | | 3.5 x10$^{-4}$ m$^3$/sec |

[1] CR χ/Q values for 24 – 96 and 96 – 720 hours are adjusted to incorporate occupancy factors

**Table 3**
**FitzPatrick Offsite Atmospheric Dispersion Factors (sec/m$^3$)**

| Receptor/Duration | | χ/Q (sec/m$^3$) |
|---|---|---|
| EAB χ/Q For evaluating releases from containment leakage and ESF leakage via SGTS | | |
| | 0 - 720 hours | 5.24 x10$^{-5}$ |
| EAB χ/Q For evaluating MSIV leakage (TB release) | | |
| | 0 - 720 hours | 1.79 x10$^{-4}$ |
| EAB Breathing rate | | 3.5 x10$^{-4}$ m$^3$/sec |
| LPZ χ/Q For evaluating releases from containment leakage and ESF leakage via SGTS | | |
| | 0 - 4 hours | 2.04 x10$^{-5}$ |
| | 4 - 8 hours | 2.17 x10$^{-6}$ |
| | 8 - 24 hours | 9.53 x10$^{-7}$ |
| | 24 - 96 hours | 3.90 x10$^{-7}$ |
| | 96 - 720 hours | 1.08 x10$^{-7}$ |
| LPZ χ/Q For evaluating MSIV leakage (TB release) | | |
| | 0 - 8 hours | 2.00 x10$^{-5}$ |
| | 8 - 24 hours | 1.34 x10$^{-5}$ |
| | 24 - 96 hours | 5.59 x10$^{-6}$ |
| | 96 - 720 hours | 1.60 x10$^{-6}$ |
| LPZ Breathing rates | | |
| | 0 - 8 hours | 3.5 x10$^{-4}$ m$^3$/sec |
| | 8 - 24 hours | 1.8 x10$^{-4}$ m$^3$/sec |
| | 24 - 720 hours | 2.3 x10$^{-4}$ m$^3$/sec |

**Table 4**
**FitzPatrick Control Room Data and Assumptions**

| | |
|---|---|
| Control room (CR) pressure boundary envelope free volume | 116,640 $ft^3$ |
| CR boundary envelope volume used in analysis | 101,000 $ft^3$ |
| CREVAS actuation time following accident initiation[1] | 30 minutes |
| CR normal operation outside air intake | 1,920 ±10% |
| CR normal operation outside air intake used in analysis | 2,112 cfm (1,920 + 10%) |
| CR normal operation exhaust flow | 1,120 ± 10% |
| CR normal operation exhaust flow used in analysis | 1,232 cfm (1,120 + 10%) |
| CR normal operation leakage – exfiltration flow rate | 800 ± 10% |
| CR normal operation leakage used in analysis | 880 cfm (800 + 10%) |
| | |
| CREVAS filtered air intake | 1,000 cfm ± 10% |
| CREVAS filtered air intake used in analysis | 900 cfm (1,000 – 10%) |
| CREVAS filtered recirculation rate | 0 cfm (no recirc. filter) |
| | |
| CR emergency mode unfiltered intake after DMPR-105 closure | |
| Measured leakage through MOV-108 | 238 cfm |
| CR emergency mode unfiltered intake used in analysis | 300 cfm |
| Unfiltered inleakage with CR isolated with | |
| No MOV failures in emergency mode | 100 cfm |
| Total unfiltered inleakage used in analysis | 400 cfm (300 + 100) |
| Total CR exfiltration after isolation[2] | 1,300 cfm |
| | |
| CR emergency mode credited intake filter efficiency | |
| Elemental iodine | 97% |
| Organic iodide | 97% |
| Particulates | 97% |
| | |
| CR operator breathing rate (0 – 720 hours) hours | 3.5 $x10^{-4}$ $m^3$/sec |
| | |
| CR occupancy factors | |
| 0 - 24 hours | 1.0 |
| 24 - 96 hours | 0.6 |
| 96 - 720 hours | 0.4 |

[1] Includes time for manual closure of 70DMPR-105
[2] Exfiltration accounts for a filtered intake of 900 cfm plus a total unfiltered intake of 400 cfm

**Table 5 (Page 1 of 2)**
**FitzPatrick Data and Assumptions for the Loss-of-Coolant Accident**

| | |
|---|---|
| Licensed power level | 2536 MWt |
| Reactor power used in analysis includes 2% margin | 2,586.72 MWt |
| | |
| Drywell volume | 154,476 $ft^3$ |
| Drywell volume used in analysis | 150,000 $ft^3$ |
| | |
| Suppression chamber (Torus) air space volume | 112,476 $ft^3$ |
| Torus air space volume used in analysis | 109,000 $ft^3$ |
| Drywell plus torus free air volume used in analysis | 259,000 $ft^3$ |
| Containment particulate natural deposition | Not credited |
| Containment elemental iodine natural deposition | Not credited |
| Containment leak rate into reactor building (RB) | |
| 0 - 24 hours | 1.5 volume percent per day |
| 24 - 720 hours | 0.75 volume percent per day |
| | |
| RB free air volume | 2,373,660 $ft^3$ |
| RB free air volume used in analysis | 2,370,000 $ft^3$ |
| RB volume credited for mixing | 50% = 1,185,000 $ft^3$ |
| RB drawdown time | 17.2 minutes |
| RB drawdown time used in analysis | 20 minutes |
| RB bypass fraction prior to RB drawdown for both drywell leakage and ESF leakage | 100% from 0 to 20 minutes |
| RB bypass after drawdown | 0% after 20 minutes |
| RB SGTS Exhaust flow rate | 6,000 cfm ±10% |
| RB SGTS Exhaust flow rate used in analysis | 6,600 cfm |
| SGTS allowable penetration | 1.5% |
| Generic Letter 99-01 safety factor | 2 |
| SGTS Charcoal and HEPA Filter efficiencies | 97% particulate and iodine species |
| | |
| Drywell spray flow rate | 5,600 gpm |
| Spray initiation time | 20 min |
| Spray termination time | 4 hours |
| Elevation of upper drywell header | 311 feet - 3 inches |
| Elevation of lower drywell header | 287 feet - 6 inches |
| Elevation of bottom drywell floor | 256 feet - 6 inches |
| Average spray fall height | 42.875 feet |
| Flow rate and fall height correction for obstructions | 0.67 (reduced by 1/3) |
| Drywell aerosol spray removal constants | 26.36 $hr^{-1}$ for DF ≤ 50<br>2.636 $hr^{-1}$ after DF = 50 |
| Drywell elemental iodine spray removal | Inventory reduced by factor of 200<br>At approximately 2.1 hours |

**Table 5 (Page 2 of 2)**
**FitzPatrick Data and Assumptions for the Loss-of-Coolant Accident**

| | |
|---|---|
| Torus water volume | 106,442 $ft^3$ |
| Torus water volume used in analysis | 106,000 $ft^3$ |
| ESF leakage rate per TS 5.5.2 | 5 gpm |
| ESF leakage used in analysis | 10 gpm |
| ESF leakage Duration | 0 to 30 days |
| ESF Flashing fraction | 10% |
| Suppression pool scrubbing | Not credited |
| Suppression pool (torus) pH | >7.0 |
| Chemical form of iodine in ESF leakage | |
| Elemental iodine | 97% |
| Organic iodide | |
| | |
| Total MSIV leak rate through all four valves | 270 scfh @ 45 psig |
| MSIV Leak rate through one line with failed MSIV | 135 scfh @ 45 psig |
| MSIV Leak rate through intact line | 135 scfh @ 45 psig |
| Main Steam pipe diameter | 24 inches |
| Main Steam wall thickness | 1.219 inches |
| Corrosion allowance for MSLs | 0.12 inches |
| Main Steam Line seismic boundary | Up to the TSVs |

B. Hanson

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 338 RE: ALTERNATIVE SOURCE TERM FOR CALCULATING LOSS-OF-COOLANT ACCIDENT DOSE CONSEQUENCES (EPID L-2019-LLA-0171) DATED JULY 21, 2020

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