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

October 8, 2020

Mr. Kevin Cimorelli
Site Vice President
Susquehanna Nuclear, LLC
769 Salem Boulevard
NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 276 AND 258 RE: REVISE TECHNICAL SPECIFICATION 5.5.2 TO MODIFY THE DESIGN-BASIS LOSS-OF-COOLANT ACCIDENT ANALYSIS (EPID L-2020-LLA-0000)

Dear Mr. Cimorelli:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 276 to Renewed Facility Operating License No. NPF-14 and Amendment No. 258 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, respectively. These amendments consist of changes to the technical specifications in response to your application dated January 2, 2020, as supplemented by letter dated June 2, 2020.

These amendments revise Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment," and modify the design-basis accident loss-of-coolant accident analysis described in the Susquehanna Updated Final Safety Analysis Report. Specifically, the changes use an updated version of the ORIGEN code, introduce a new source term to account for the introduction of ATRIUM 11 fuel, use new assumptions that decrease the assumed emergency safety feature leakage into secondary containment, increase the assumed maximum allowable standby gas treatment system exhaust flow rate from secondary containment, and increase the allowed control structure unfiltered in-leakage that is assumed in the design-basis accident loss-of-coolant accident analysis.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* Notice.

Sincerely,

/RA/

Sujata Goetz, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 276 to
   License No. NPF-14
2. Amendment No. 258 to
   License No. NPF-22
3. Safety Evaluation

cc: Listserv

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

SUSQUEHANNA NUCLEAR, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 276
Renewed License No. NPF-14

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

   A. The application for the amendment filed by Susquehanna Nuclear, LLC, dated January 2, 2020, as supplemented by letter dated June 2, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-14 is hereby amended to read as follows:

    (2) <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2020.10.08 12:54:35 -04'00'

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

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

Date of Issuance: October 8, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 276

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

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	Insert
Page 3	Page 3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
5.0-8	5.0-8

(3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, 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 license shall be deemed to contain and is subject to the conditions specified in the Commission’s regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(36), 2.C.(37), 2.C.(38), and 2.C.(39) to this license shall be completed as specified.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 276, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 178 to Facility Operating License No. NPF-14, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 178. For SRs that existed prior to Amendment 178, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 178.

5.5 Programs and Manuals

5.5.1 (ODCM) (continued)

shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Cleanup, Standby Gas Treatment, Post Accident Sampling (until such time as a modification eliminates the PASS penetration as a potential leakage path) and Containment Air Monitoring Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

SUSQUEHANNA NUCLEAR, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. NPF-22

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

   A. The application for the amendment filed by Susquehanna Nuclear, LLC, dated January 2, 2020, as supplemented by letter dated June 2, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-22 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2020.10.08 12:55:14 -04'00'

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

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

Date of Issuance: October 8, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 258

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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	Insert
Page 3	Page 3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
5.0-8	5.0-8

(3) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, posses, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, posses, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, 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 license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission nor or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(20), 2.C.(21), 2.C.(22), and 2.C.(23) to this license shall be completed as specified.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151 to Facility Operating License No. NPF-22, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 151. For SRs that existed prior to Amendment 151, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 151.

5.5 Programs and Manuals

5.5.1 ODCM (continued)

shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Cleanup, Standby Gas Treatment, Post Accident Sampling (until such time as a modification eliminates the PASS penetration as a potential leakage path) and Containment Air Monitoring Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Not Used

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 276 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 258 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-22

SUSQUEHANNA NUCLEAR, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By application dated January 2, 2020 (Agencywide Documents Access and Management system (ADAMS) Accession No. ML20002B254), as supplemented by letter dated June 2, 2020 (ADAMS Accession No. ML ML20155K659), Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request for changes to the Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, Technical Specifications (TSs).

The supplement dated June 2, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 5, 2020 (85 FR 26731).

The amendments would revise TS 5.5.2, "Primary Coolant Sources Outside Containment," and modify the design-basis accident (DBA) loss-of-coolant accident (LOCA) analysis described in the Susquehanna Updated Final Safety Analysis Report (UFSAR). Specifically, the changes would use an updated version of the ORIGEN code, introduce a new source term to account for the introduction of ATRIUM 11 fuel, use new assumptions that decrease the assumed emergency safety feature (ESF) leakage into secondary containment, increase the assumed maximum allowable standby gas treatment system exhaust flow rate from secondary containment, and increase the allowed control structure unfiltered inleakage that is assumed in the DBA LOCA analysis.

2.0 REGULATORY EVALUATION

The changes would include accident source term calculation methodology and the LOCA parameters for postulated accident analysis, including the inputs and assumptions used in the Susquehanna LOCA dose consequences analysis. The proposed changes would support loading ATRIUM 11 fuel and would increase the margin for surveillance testing.

The NRC staff considered the following regulatory requirements and guidance in its review of the LAR.

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67(b)(2), "Accident source term," provides that the NRC may issue a license amendment for a licensee who seeks to revise its current accident source term in design-basis radiological consequence analyses only if the applicant's analysis demonstrates with reasonable assurance that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 [roentgen equivalent man] rem) total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Section 50.36(c)(3) of 10 CFR states:

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

Section 50.36(c)(5) of 10 CFR states, in part:

> Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants" (GDC), Criterion 19, "Control room," states:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy

> of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Regulatory Guide (RG) 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”, dated July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. Additionally, RG 1.183 provides that ESF systems that recirculate sump water outside of primary containment are assumed to leak during their intended operation.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” (SRP) Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, dated July 2000 (ADAMS Accession No. ML003734190), provides review guidance to the NRC staff for the review of AST amendment requests. Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for the fuel handling accident are a total effective dose equivalent (TEDE) of 6.3 rem at the exclusion area boundary for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone, and 5 rem in the control room for the duration of the accident.

NRC Regulatory Issue Summary 2006-04, “Experience with Implementation of Alternative Source Terms,” dated March 7, 2006 (ADAMS Accession No. ML053460347), discusses experiences with analyzing an accident involving a release from off-gas or waste systems.

License Amendment Nos. 239 and 216 for Susquehanna, Units 1 and 2, dated January 31, 2007 (ADAMS Accession No. ML070080301), incorporated a full-scope application of an AST methodology in accordance with 10 CFR 50.67.

2.1 Proposed Technical Specification Changes

The proposed amendments would modify TS 5.5.2 to remove the term "Scram Discharge" from the systems included in the program. The proposed amendment also modifies the Bases for TS 3.6.4.1, "Secondary Containment."

Specifically, the licensee states the proposed changes would:

- use an updated version of the ORIGEN code,
- introduce a new source term to account for the introduction of ATRIUM 11 fuel,
- use new inputs/assumptions that decrease the assumed Emergency Safety Feature leakage into secondary containment,
- increase the assumed maximum allowable Standby Gas Treatment System exhaust flow rate from secondary containment, and
- increase the allowed control structure unfiltered in leakage that is assumed in the DBA LOCA dose analysis.

3.0 TECHNICAL EVALUATION

UFSAR Chapter 15 describes the postulated DBA and transient scenarios applicable to Susquehanna during power operations. These analyses demonstrate that the plant could be operated safely and that radiological consequences from postulated accidents do not exceed the regulatory guidelines of 10 CFR 50.67 or 10 CFR Part 100, as applicable. Two basic groups of events are pertinent to safety, which are abnormal operational transients and postulated DBAs; these two groups were investigated separately. The analyses of the abnormal operational transients evaluate the ability of the plant protection features to ensure that, during these transients, no fuel damage occurs, and the reactor coolant system pressure limit is not exceeded. The safety design limits require that damage to the fuel be limited and that no nuclear system process barrier damage results from any abnormal operational occurrence. Thus, analysis of this group of events evaluates the features that protect the first two radioactive material barriers. Analysis of the events in the second group, postulated DBAs, evaluates situations that require functioning of the engineered safeguards in order to protect the fission product barriers, including containment, in order to minimize the offsite radiological consequences.

3.1 Loss-of-Coolant Accident

UFSAR Section 15.6.5 describes a loss of coolant accident (LOCA). This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coupled with severe natural environmental conditions, including earthquake coincidence. This general scenario does not represent any specific accident sequence but is representative of a class of severe damage incidents that were evaluated in the development of the 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.

In the evaluation of the LOCA design-basis radiological analysis, the licensee's dose contributions include the following activity release pathways:

- primary containment leakage to the reactor building (RB),
- primary containment bypass leakage directly to the environment,
- ESF leakage to the RB, and
- main steam isolation valve (MSIV) leakage to the environment via the condenser.

The analysis also included the following DBA LOCA dose contributors to the control room habitability analysis:

- contamination of the control room atmosphere by released activity,
- shine from containment, RB, and turbine building, and
- shine from piping, components, and control room filter loading.

3.1.1 Accident Source Term – Fission Product Inventory

RG 1.183, Regulatory Position 3.1, provides methods and assumptions acceptable to the NRC staff to define the fission product inventory for an AST. The currently approved Susquehanna accident source term was developed for cores with ATRIUM 10 fuel using SAS2H/ORIGEN-S from SCALE 4.4.a. The SAS2H/ORIGEN-S code is comprised of an advanced version of ORIGEN and is consistent with the source term code recommendations given by RG 1.183 for generation of ASTs. The current ORIGEN-S based core inventory uses an ATRIUM 10 fuel cycle with core thermal power of 4,032 megawatt thermal (MWt) (102 percent of the Susquehanna rated thermal power of 3,952 MWt) and core average burnup of 39 gigawatt-days per metric ton of uranium (GWd/MTU).

Consistent with RG 1.183, Regulatory Position 3.1, the proposed ATRIUM 11 core source term was developed using TRITON/ORIGEN-ARP from SCALE 6.2.3. ATRIUM 11 has a higher assembly uranium mass and higher core average burnup than ATRIUM 10, resulting in a conservative core source term for Susquehanna core designs. The core inventory uses an ATRIUM 11 fuel cycle with core thermal power of 4,032 MWt and core average burnup of 41 GWd/MTU. The source term assumes conservative fuel uranium mass and enrichment sensitivities performed to bound the range of expected U-235 enrichments in ATRIUM 11 reload fuel. The NRC staff finds that the use of the updated version of the ORIGEN code is acceptable.

3.1.2 Emergency Safety Feature System Leakage

RG 1.183, Appendix A, Regulatory Position 5, identifies ESF system leakage as a source of activity release for evaluating the radiological consequences of the design-basis LOCA. The source of ESF system leakage is recirculated sump water outside of the primary containment that is assumed to leak during the intended period of operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems.

The Susquehanna current licensing basis considers two sources of potential ESF system leakage in the release model, which assumes a total of 20 gallons per minute (gpm) of flow. The first source evaluated is ESF system leakage directly into secondary containment. In addition to the 5 gpm ESF system leakage, a non-ESF system leakage of 15 gpm is assumed for suppression pool water leakage into secondary containment. This leakage was conservatively assumed to bound any leakage that could occur from other sources inside secondary containment, including the control rod drive (CRD) insert/withdrawal lines and the scram discharge volume (SDV).

Those ESF systems that contribute to this leakage are identified in UFSAR Section 18.1.69 and the leakage rate test program. The total leakage from these systems is maintained ≤ 2.5 gpm in accordance with TS 5.5.2 and UFSAR Section 18.1.69. The dose analysis assumes a value of 5 gpm, which is at least two times the acceptance criteria for the sum of the simultaneous leakage from all components in the ESF recirculation systems. The non-ESF system leakage paths described in UFSAR Section 6.2.4.3.2.3 also contribute to the amount of post-accident liquid leakage from primary to secondary containment. As discussed by the licensee, this assumed non-ESF leakage has been assumed to conservatively bound the total post-accident liquid leakage for dose analysis purposes.

The proposed change removes the additional non-ESF leakage from the flow path so that the LOCA analysis will use a total of 5 gpm instead of 20 gpm of flow through the ESF leakage pathway. The revision to ESF leakage pathways results in changes to UFSAR Sections 6.2.4.3.2.3, 15.6.5.5.1.2, and 18.1.69.3, as well as UFSAR Tables 6.2-22 and 15.6-22. Additionally, the revision to ESF leakage pathways results in eliminating scram discharge from the list of systems within the scope of the primary coolant sources outside containment program in TS 5.5.2. The licensee justifies, in part, the removal of the non-ESF leakage from the ESF leakage pathway because is it not required by RG 1.183, Appendix A. In the LAR, the licensee states:

> … a leakage of 15 gpm was assumed for suppression pool water leakage into secondary containment. This leakage was conservatively assumed to bound any leakage which could occur from other sources inside secondary containment including the Control Rod Drive (CRD) insert/withdrawal lines and the Scram Discharge Volume (SDV). This additional leakage is not ESF System leakage, is not required by RG 1.183, Appendix A, and was strictly included as a conservative assumption. Therefore, a total water leakage of 20 gpm was conservatively assumed for the DBA LOCA dose analysis for the initial AST submittal.

Since the non-ESF leakage paths contribute to the amount of post-accident liquid leakage from the primary to secondary containment, the licensee provided additional information expanding on the technical basis of removing these leakage flow paths from the current licensing basis dose consequence analysis.

The SDV is part of the CRD insert/withdrawal lines. When actuated, control rod motion is caused by movement of a drive piston internal to the drive mechanism. The motive force for movement of the drive piston is pressurized water on either side of the drive piston. The underside of the piston receives pressurized water from the hydraulic control unit during a scram. The overside of the drive piston is filled with normal reactor coolant prior to a postulated scram. During the scram, the pressurized hydraulic control unit water provides the motive force for the upward movement of the drive piston, which displaces reactor coolant on the overside of the piston into the SDV. The SDV is vented during normal plant operation and is isolated on a scram signal. Therefore, the activity in the SDV post-LOCA is equivalent to the reactor coolant activity during normal operation and would represent a very small fraction to the total computed radiological dose.

The CRD insert and withdrawal lines are the pipes connected to the containment penetrations associated with the flow to the CRD mechanism (e.g., CRD charging water, CRD water header, CRD cooling water header) and the return flow (e.g., CRD exhaust water, SDV). UFSAR Section 6.2 describes these lines as follows:

> The CRD system insert and withdraw lines penetrate the drywell. The CRD insert and withdrawal lines are not part of the reactor coolant pressure boundary since they do not directly communicate with the reactor coolant. The classification of these lines is quality group B, and they are designed in accordance with ASME [American Society of Mechanical Engineers] Section III, Class 2. The basis on which the CRD insert and withdrawal lines are designed is commensurate with the safety importance of maintaining the pressure integrity of these lines.
>
> It has been accepted practice not to provide automatic isolation valves for the CRD insert and withdrawal lines to preclude a possible failure mechanism of the scram function. The CRD insert and withdrawal lines can be isolated by the solenoid valves outside the primary containment. The lines that extend outside the primary containment are small and terminate in a system that is designed to prevent out-leakage. Solenoid valves are normally closed, but open on rod movement and during reactor scram. In addition, a ball check valve located in the CRD flange housing automatically seals the insert line in the event of a break. Finally, manual shutoff valves are provided.

The CRD insert and withdrawal lines are filled with water from the condensate storage tank during normal operation. This is clean water from a radioactivity standpoint. The only change to the water in these systems post-scram is the SDV discussed above. A review of NUREG-1433, Revision 4, “Standard Technical Specifications – General Electric (BWR/4),” determined that the Standard Technical Specifications do not include the SDV in TS 5.5.2, “Primary Coolant Sources Outside Containment.” TS 5.5.2 minimizes leakage from systems outside primary containment that contain highly radioactive fluid after a transient or accident. As discussed above, the CRD insert and withdrawal lines and the SDV will not contain highly radioactive fluid post-LOCA. Therefore, the NRC staff support the licensee’s conclusion that the leakage from these systems need not be included in the dose consequence analysis or included in TS 5.5.2. This change aligns with accepted consequence modeling positions of RG 1.183.

The revision to ESF leakage pathways supports eliminating scram discharge from the list of systems within the scope of the “Primary Coolant Sources Outside Containment Program” in TS 5.5.2. Based on the adequacy of the ESF system leakage evaluation described above, the staff finds that the corresponding changes in the TSs are acceptable and continue to meet the regulations in 10 CFR 50.36(c)(5).

3.1.3 Secondary Containment Inleakage

The secondary containment structure completely encloses the primary containment structure such that a dual-containment design is utilized to minimize the spread of radioactivity to the environment. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary into secondary containment following a DBA. The secondary containment boundary consists of the RB structure and associated removable walls and panels, hatches, doors, dampers, sealed penetrations, and valves. The secondary containment is divided into Zone I, Zone II, and Zone III, each of which must be operable, depending on plant status and the alignment of the secondary containment boundary.

Zones I and II are the portions of the RB below elevation 779'-1" surrounding the Units 1 and 2 primary containments, respectively. Zone III consists of the portion of the reactor buildings above elevation 779'-1" with the exception of the heating, ventilation, and air conditioning equipment rooms, which are not part of the secondary containment. The Unit 1 secondary containment boundary can be modified to exclude Zone II. Similarly, the Unit 2 secondary containment boundary can be modified to exclude Zone I.

The proposed change would increase the allowable inleakage from 140 percent per day to 225 percent per day based on the most recent testing. The inleakage value for the most limiting configuration (Zones II and III with the railroad bay aligned to secondary containment) was 3,630 cubic feet per minute (cfm) compared to a limit of 4,000 cfm. This change is proposed in order to provide additional margin for secondary containment drawdown testing performed at Susquehanna per Surveillance Requirement (SR) 3.6.4.1.5.

The proposed change is within the design capacity of the standby gas treatment (SGT) system. Therefore, the SGT current licensing basis system equipment is not impacted due to the increased flow rate. One result of the proposed change is the ability of the SGT system to draw down the RB to a vacuum of 0.25-inch water gauge post-accident. Per SR 3.6.4.1.4, the secondary containment 0.25-inch water gauge vacuum must be reestablished within 5 minutes, which is bounded by the 10-minute drawdown assumed in the dose analysis. Therefore, the drawdown time will continue to be validated through SR 3.6.4.1.4.

3.1.4 Control Room Habitability Envelope Unfiltered Inleakage

Under accident conditions, habitability for the control room habitability envelope (CRHE) is provided by the control room emergency outside air supply system. This system provides habitability zone isolation and a positive pressure for the CRHE. Consistent with the assumptions of the AST, this occurs in sequence with the automatic initiation of the SGT system prior to commencement of a significant release of radioactive material to the environment.

The control structure unfiltered inleakage is leakage into the control structure habitability boundary that bypasses the emergency filtration system (control room emergency outside air supply system). The DBA LOCA analysis assumes a value of control structure unfiltered inleakage that is controlled by TS 5.5.14, “Control Room Envelope Habitability Program,” and verified by SR 3.7.3.4. The proposed change would increase the allowable unidentified unfiltered inleakage from 500 cfm to 600 cfm in order to provide additional margin for the tracer gas inleakage tests performed at Susquehanna per SR 3.7.3.4, as required by TS 5.5.14. The envelope at Susquehanna encompasses multiple floor elevations in the control structure. In order to pressurize a boundary of this size, a substantial amount of outside air is required (5,810 cfm). Substantial error is introduced into the testing due to measurement of this flow rate and also due to maintaining a uniform concentration in the various rooms on different elevations. In the last tracer gas testing performed in 2017, uncertainty associated with the ‘A’ train was ±326 cfm.

Per Susquehanna, Units 1 and 2, TSs 3.7.3.4 and 5.5.14, the (control room emergency outside air supply system) filtered intake flow ranges from 5,229 cfm to 6,391 cfm. The licensee evaluated the LOCA for filtered intake flows of 5,229 cfm and 6,391 cfm and determined that the 5,229 cfm flow rate was limiting. The proposed LOCA CRHE analysis assumes that 600 cfm of unfiltered inleakage exists, which bounds the tracer gas testing results.

The staff reviewed the information above and finds the proposed change of the allowable unfiltered inleakage from 500 to 600 cfm acceptable because it is based on performance-based, plant-specific operational experience, and it is controlled by TS 5.5.14, "Control Room Envelope Habitability Program," per 10 CFR 50.36(c)(5) and verified by SR 3.7.3.4 per 10 CFR 50.36(c)(3).

3.1.5 Control Room Habitability Envelope Continuous Occupancy Locations

The design basis for the CRHE is to provide adequate radiation protection to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. It is generally understood that an objective of the criteria is to ensure that the design of the control room and its habitability systems is such that a "shirt-sleeved" environment would be provided for the control room operators. Such an environment is perceived to be supportive of facilitating operator response to normal and accident conditions. Another objective is to ensure that the radiation dose levels in the control room would make it the "safest" location on site, thereby allowing the operators to remain in the control room and not evacuate. Any reduction in the ability of the operators to respond appropriately during an accident is properly viewed as having a potential impact on the public health and safety.

Under accident conditions, radiation doses to control room personnel may result from several sources. While in the control room, personnel are exposed to beta and gamma radiation from gaseous fission products that enter after an accident via the ventilation system or from unfiltered air entering the control room. In addition, personnel may be subject to gamma shine dose from fission products in the containment and RB from contained system sources and from fission products in the atmosphere outside the control room.

RG 1.196, Revision 1, "Control Room Habitability at Light-Water Nuclear Power Reactors," dated January 2007 (ADAMS Accession No. ML063560144), defines the control room envelope as the plant area defined in the facility licensing basis that, in the event of an emergency, can be isolated from the plant areas and the environment external to the control room envelope. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.

Regulatory Position 4.2.6 of RG 1.183 discusses dose receptor for these locations to be the hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 to 30 days.

The Susquehanna CRHE is designed with adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of specified design values. Although all areas of the control structure envelope are designed for habitability following a DBA LOCA, not all areas inside the envelope require continuous occupancy.

The Susquehanna, Units 1 and 2, CRHEs are defined in UFSAR Figure 6.4-1A. Of these areas within the CRHE, the following areas are currently assumed to be continuously occupied following a DBA LOCA:

- Control Structure Elevation 729'-1" (all areas), which contains the
  - Main Control Room (Room C-409),
  - Shift Technical Advisor Office (Room C-401),
  - Backup Operation Support Center (Room C-402), and
  - Other ancillary rooms.
- Control Structure Elevation 741'-1" (all areas), which contains the
  - Technical Support Center (Room C-410),
  - Electrical Equipment Room (Room C-413), and
  - NRC Conference Room (Room C-414).
- Computer Room (Room C-202) at Elevation 698'-0"

As approved by Susquehanna, Units 1 and 2, License Amendment Nos. 239 and 216, respectively, only the following areas within the CRHE are required to meet the continuously occupied post-accident guidance of RG 1.183, Section 4.2.6:

- Main Control Room (Room C-409), Elevation 729'-1"
- Technical Support Center (Room C-410), Elevation 741'-1", and
- Computer Room (Room C-202), Elevation 698'-0".

The post-accident locations that do not require continuous occupancy for safe plant operation or for execution of the emergency plan include:

- Shift Technical Advisor Office (Room C-401), Elevation 729'-1",
- Backup Operation Support Center (Room C-402), Elevation 729'-1",
- Electrical Equipment Room (Room C-413), 741'-1", and
- NRC Conference Room (Room C-414), 741'-1".

The maximum dose rate inside these areas occurs at the closest location to the 14" GBB-101 core spray line. The shine dose rate from this contained source for corresponding locations in the north and south ends of these areas is the same due to the same source locations in both Susquehanna, Units 1 and 2, RBs. As a result of the location of the contained source piping, the electrical equipment room (C-413) has the limiting dose of 4.69 rem TEDE.

Based on its review, the NRC staff agrees with the licensee's determination that based on a conservative analysis of the direct shine from core spray piping located in the adjacent RB, access to these certain ancillary rooms can be restricted based on radiation monitoring. The NRC staff also agrees with the licensee's assertion that the access controls described will not impact the performance of critical safety functions or affect the ability of the control room operators to perform the tasks necessary for a safe shutdown. Therefore, the limited access controls within the CRHE meet the requirements of GDC 19 and 10 CFR 50.67 and are acceptable to the NRC staff.

Technical Conclusion

The NRC staff performed confirmatory analyses and reviewed the assumptions, inputs (see Table 1 below), and methods used by the licensee to assess the radiological impacts of the proposed changes and finds that the licensee's proposed changes use analysis methods and assumptions consistent with the guidance contained in RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable criteria. The NRC staff finds, with reasonable assurance, that Susquehanna, as modified by the proposed changes, will continue to provide enough safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters. The NRC staff concludes, with reasonable assurance, that the licensee has demonstrated that the dose consequences for postulated accidents at Susquehanna would not exceed the dose acceptance criteria specified in 10 CFR 50.67 and RG 1.183. Therefore, the NRC staff finds the proposed changes to be acceptable from a dose consequence perspective.

Table 1: Input Parameters for the Susquehanna Design-Basis Loss-of-Coolant Accident

Parameters	Revised Design Basis
Core Thermal Power Level	4,032 MWt
Activity Inventory in Core Ci/MWt	60 dose-significant isotopes used in RADTRAD
Radioisotope Decay Properties	RADTRAD Table 1.4.3.2-3
Activity Release to Containment	Per RG 1.183, Table 1
	(Gap and early in-vessel phases only)
Release Timing	Per RG 1.183, Table 4
Radioiodine Chemical Species	95% aerosol (CsI) [cesium iodine]
	4.85% elemental
	0.15% organic
Primary Containment Volume	Drywell free volume = 239,600 ft^3 Wetwell free volume = 148,590 ft^3 Total free volume = 388,190 ft^3
Primary Containment Cleanup (Natural Deposition)	Aerosol removal via natural deposition
	(10^{th} percentile Powers' Model)
Primary Containment Cleanup (Drywell Sprays)	No credit taken
Primary Containment Design Leak Rate	1%/day for first 24 hours, and 0.5%/day thereafter
Primary Containment Design Leak Rate into Secondary Containment (RB)	0.9777%/day for first 24 hours, and 0.4889%/day thereafter
Containment Bypass Leak Rate	0.0223/day (0.0601 cfm) 0 – 24 hours

Parameters	Revised Design Basis
	Hours
	0.01115%/day (0.03005 cfm)
	Thereafter
Offsite Breathing Rates(m^3/seconds (sec))	3.5E-04, 0-8 hours
	1.8E-04, 8-24 hours
	2.3E-04, 1-30 day
Dose Conversion Factors	RADTRAD Table 1.4.3.3-2
Suppression Pool Scrubbing	Not credited
Secondary Containment (RB) Free Air Volume	Zone I 1,488,600 ft^3 Zone II 1,598,600 ft^3 Zone III 2,668,000 ft^3
	Total Volume = 5,755,200 ft^3
Secondary Containment Volume Mixing Fraction/Analysis Volume	50% mixing
	RADTRAD Volume = 2,078,300 ft^3
SGTS at Full Flow Post-LOCA	30 seconds
Post-LOCA RB Drawdown Time	10 minutes (used in analysis)
RB Leakage Till End of Drawdown	225%/day or 6494.7 cfm for two Zone (I and III) mixing
RB Leakage After Drawdown	0 cfm
SGTS Flow Rate	11,110 cfm for the first 10 minutes and
	6,494.7 cfm thereafter
SGTS Filter Bed Depth	8 inches (in.) charcoal
SGTS Filter Bed Efficiency	99 for all iodine species
MSIV Leak Rate	300 standard cubic feet per hour (scfh) (four lines) modeled as:
	100 scfh in one assumed faulted line
	66.67 scfh each in the remaining lines
Main Steam Line (MSL) “C” Faulted Line Length to Condenser	NA pipe not used for plateout
MSL “A” Length/ IDs MSL “B” Length/ IDs MSL “D” Length/ IDs	20.5625 feet (ft.)/23.647" + 2.5 ft./22.062"
	22.5625 ft./23.647" + 2.5 ft./22.062"
	20.5625 ft./23.647" + 2.5 ft./22.062"

Parameters	Revised Design Basis
MSL Volume – Reactor to Condenser	MSL "A" – 67.8 ft^3 MSL "B" – 73.9 ft^3 MSL "D" – 67.8 ft^3
MSL Projected Internal Surface Area – For Aerosol Plateout	MSL "A" – 22.1 ft^2 MSL "B" – 24.0 ft^2 MSL "D" – 22.1 ft^2
MSL Internal Surface Area – For Elemental Plateout	MSL "A" – 136.6 ft^2 MSL "B" – 151.0 ft^2 MSL "D" – 136.6 ft^2
Leakage Split Between Drain Line and MSL/High-Pressure Turbine (HPT) Pathways to Condenser	98.7%, drain line 1.3%, MSL/HPT
RADTRAD model for Leakage Split Between Drain Line and MSL/HPT Pathways to Condenser	100%, drain line
Minimum Drain Line Length Volume	281.2 ft – 18.828 ft^3 – 40.688 ft^2
Surface Area for Aerosol Plateout Surface Area for Elemental Plateout	255.65 ft^2
Effective Aerosol and Elemental Removal for MSIV Path	Table 6 of EC-RADN-1125 (unchanged)
Effective Condenser Volume for Each Pathway	98,601 ft^3
Effective Removal Efficiency in Condenser for Drain Line Pathway	99.6%, effective on aerosols and elemental iodine No organic iodine removal
Modeled MSIV Leakage – MSL Inlet, Initial Pressure, and Temperature	Drywell peak accident values MSL pressure = 50 pounds per square inch absolute (psia) MSL temperature = 340 degrees Fahrenheit (°F)
MSIV Leakage Pressure and Temperature into Condenser	Pressure = 1 standard atmosphere (atm), Temperature = 100 °F
MSIV Source Release Timing	Instantaneous
Minimum Suppression Pool Volume Post-LOCA	122,410 ft^3 (low volume based on 22 ft. pool level) 610,000 lbm (reactor water mass) 610,000 pound-mass (lbm)/62.4 lb/ft^3 = 9,776 ft^3 Total: 132,000 ft^3
ESF System Leakage Source Term to Environment	Iodine only
ESF Leakage into RB	5 gpm or 0.668 cfm

Parameters	Revised Design Basis
ESF Leakage Outside of the RB	None
ESF Leakage Post-LOCA Time	Begins at 0 seconds - Ends at 30 days
Susquehanna Post-LOCA Suppression Pool maximum temperature	< 212 °F
ESF Flash Fraction	10%
SP Iodine Species	97% elemental 3% organic
Iodine Re-evolution	None assumed since pH > 7
RB Sump Iodine Species	97% elemental 3% organic
Control Structure Habitability Envelope Total Volume	518,000 ft^3
Control Room Free Air Volume	110,000 ft^3
Geometry Correction Factor (GF)	$GF = 1,173/V^{0.338}$
	Volume (V) = CRHE or control room volume
Control Room Isolation Time	0
Emergency Intake Airflow, Total into Control Structure	5,229 cfm
Unfiltered Air Inleakage Ingress/Egress	10 cfm
Other Unfiltered Air Inleakage	600 cfm
Control Room Exhaust Flow	5,839 cfm
Emergency Filter Bed Depth	4 in. charcoal
Emergency Filter Bed Removal Efficiency	99%
Operator Breathing Rates	3.5E-04 m^3/sec (0 - 30 days)
Offsite Breathing Rates	3.5E-04 m^3/sec (0 - 8 hours)
	1.8E-04 m^3/sec (8 - 24 hours)
	2.3E-04 m^3/sec (1 - 30 days)
Operator Occupancy Factors	1.0 0-24 hours (hrs)
	0.6 1-4 days
	0.4 4-30 days
χ/Q	Exclusion area boundary:

Parameters	Revised Design Basis
	(0 - 2) 8.3E-04 sec/m^3
	Low Population Zone:
	(0 - 8 hrs) 4.9E-05 sec/m^3
	(8 - 24 hrs) 3.50E-05 sec/m^3
	(24 - 96 hrs) 1.70E-05 sec/m^3
	(96 - 720 hrs) 6.10E-06 sec/m^3
	CRHE χ/Q values listed in Table 15 of EC-RADN-1125

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on June 30, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes SRs. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (85 FR 26731). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

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

Principal Contributors: E. Dickson
C. Li

Date: October 8, 2020

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 276 AND 258 RE: REVISE TECHNICAL SPECIFICATION 5.5.2 TO MODIFY THE DESIGN-BASIS LOSS-OF-COOLANT ACCIDENT ANALYSIS (EPID L-2020-LLA-0000) DATED OCTOBER 8, 2020

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OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LA	NRR/DSS/SCPB/BC	NRR/DRA/ARCB/BC
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DATE	07/23/2020	07/22/2020	06/04/2020	06/26/2020
OFFICE	NRR/DRR/STSB/BC	OGC – NLO	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM
NAME	VCusumano	AGhoshNaber	JDanna	SGoetz
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