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

October 20, 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: NINE MILE POINT NUCLEAR STATION, UNIT 2 - ISSUANCE OF AMENDMENT NO. 182 TO CHANGE ALLOWABLE MAIN STEAM ISOLATION VALVE LEAK RATES (EPID L-2019-LLA-0115)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 182 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2). The amendment consists of changes to the technical specifications in response to your application dated May 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19151A537), as supplemented by letters dated November 21, 2019, and May 14, 2020 (ADAMS Accession Nos. ML19325D201 and ML20135G951, respectively).

The amendment revises the Nine Mile Point 2 alternative source term loss-of-coolant accident radiological analysis, combines the delayed drywell leakage and drywell leakage surveillance requirements, and changes the allowable main steam isolation valve leakage rate.

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/

Michael L. Marshall, Jr., Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 182 to NPF-69
2. Safety Evaluation

cc: Listserv

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

NINE MILE POINT NUCLEAR STATION, LLC

LONG ISLAND LIGHTING COMPANY

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 182
Renewed License No. NPF-69

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated May 31, 2019, as supplemented by letters dated November 21, 2019, and May 14, 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. NPF-69 is hereby amended to read as follows:

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

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 182, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G.
Danna

Digitally signed by James G. Danna
Date: 2020.10.20 07:57:33 -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 20, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 182

NINE MILE POINT NUCLEAR STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

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

Remove Page	Insert Page
4	4

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
3.6.1.3-12	3.6.1.3-12
3.6.1.3-13	3.6.1.3-13

(1) Maximum Power Level

Exelon Generation is authorized to operate the facility at reactor core power levels not in excess of 3988 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 182, are hereby incorporated into this license. Exelon Generation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fuel Storage and Handling (Section 9.1.SSER 4)*

a. Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three containers high.

b. When not in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility.

c. The above three fuel assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

d. The New Fuel Storage Vault shall have no more than ten fresh fuel assemblies uncovered at any one time.

(4) Turbine System Maintenance Program (Section 3.5.1.3.10 SER)

The operating licensee shall submit for NRC approval by October 31, 1989, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities. (Submitted by NMPC letter dated October 30, 1989 from C.D. Terry and approved by NRC letter dated March 15, 1990 from Robert Martin to Mr. Lawrence Burkhardt, III).

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the license condition is discussed.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Perform leakage rate testing for each primary containment purge valve with resilient seals.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 92 days after opening the valve
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.8	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 3.6.1.3.9	Verify a representative sample of reactor instrumentation line EFCVs actuates to the isolation position on an actual or simulated instrument line break signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.11	Verify the leakage rate for the secondary containment bypass leakage when pressurized to ≥ 40 psig is: a. Bypass (Drywell): ≤ 36.88 SCFH; and b. Bypass (Suppression Chamber): ≤ 1.66 SCFH.	In accordance with 10 CFR 50 Appendix J Testing Program Plan

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.12	Verify leakage rate through each MSIV is ≤ 50 scfh when tested at ≥ 40 psig.	In accordance with 10 CFR 50 Appendix J Testing Program Plan
SR 3.6.1.3.13	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with 10 CFR 50 Appendix J Testing Program Plan

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-69

NINE MILE POINT NUCLEAR STATION, LLC

LONG ISLAND LIGHTING COMPANY

EXELON GENERATION COMPANY, LLC.

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated May 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19151A537), as supplemented by letters dated November 21, 2019, and May 14, 2020 (ADAMS Accession Nos. ML19325D201 and ML20135G951, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for changes to the Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2), Technical Specifications (TSs). The requested changes would combine the delayed drywell leakage and drywell leakage surveillance requirements (SRs) and change the allowable main steam isolation valve (MSIV) leakage rate in a different SR. Additionally, the LAR proposed to revise to the Nine Mile Point 2 alternative source term (AST) loss-of-coolant accident (LOCA) radiological analysis. The proposed changes to the SRs are based on the revised AST LOCA radiological analysis and revised radiological analysis for environmental qualification (EQ).

The supplemental letters dated November 21, 2019, and May 14, 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, the Commission) staff's initial proposed no significant hazards consideration determination noticed in the *Federal Register* on September 10, 2019 (84 FR 47547).

In the LAR, the licensee stated that the refurbishment of MSIVs under the current TS requirements is a man-hour intensive effort that results in a significant cumulative worker radiation dose and expenditure of resources. Additionally, the licensee stated that increasing the MSIV leakage rate limit would significantly reduce the amount of rework on the MSIVs and would lower personnel radiation exposure by reducing the number of maintenance activities.

2.0 REGULATORY EVALUATION

2.1 System Descriptions

2.1.1 Main Steam Isolation Valves

The four main steam lines (MSLs) that penetrate the drywell are automatically isolated by the MSIVs. There are two MSIVs on each MSL—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 to bypass secondary containment and enter the environment a s a ground-level release.

2.1.2 Residual Heat Removal Drywell Sprays

The residual heat removal (RHR) drywell spray system provides overpressure protection to the primary containment by quenching steam released to the drywell during a LOCA. Each of the two RHR drywell spray subsystems contains two pumps, one heat exchanger, drywell spray valves, and a spray header in the drywell. During drywell spray operation, each RHR drywell spray subsystem recirculates water from the RHR suppression pool through an RHR heat exchanger and the RHR drywell spray nozzles. Drywell spray reduces drywell temperature and pressure through the combined effects of evaporative and convective cooling and is used to wash or scrub inorganic iodine and particulates from the drywell atmosphere into the suppression pool. The drywell spray is used during a LOCA for both the scrubbing function as well as the temperature and pressure reduction effects.

2.1.3 Primary Containment Isolation Valves

The function of the primary containment isolation valves (PCIVs) (which include the MSIVs) in combination with other accident mitigation systems is to limit fission product release during and following postulated design-basis accidents (DBAs) to within the limits of Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.67, “Accident source term.” Primary containment isolation within the time limits specified for those PCIVs designed to close automatically and within the TS leakage limits ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA. According to the Nine Mile Point 2 TS Bases B.3.6.1.3, Revision 0, “PCIVs,” “Applicable Safety Analyses,” the safety analysis of any event requiring isolation of primary containment is applicable to Limiting Condition for Operation (LCO) 3.6.1.3 and the “DBAs that result in a release for which the consequences are mitigated by PCIVs are a LOCA and a main steam line break (MSLB).”

2.1.4 Standby Liquid Control System

In boiling-water reactors (BWRs) such as Nine Mile Point 2, the standby liquid control system (SLCS) is required to mitigate an anticipated transient 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 SLCS was designed as a backup method to maintain the reactor subcritical without control rods after an ATWS. Sodium pentaborate solution is stored in the SLCS tank and injected directly into the lower plenum of the reactor pressure vessel (RPV) as a means to shut down the reactor following an ATWS.

2.1.5 Control Room Envelope Filtration System

The control room envelope filtration (CREF) system plays a role in the AST dose calculations. However, input value changes have no impact on the TSs governing the CREF system. The CREF train filters a combination of incoming air from outside and recirculation flow from the control room through high-efficiency particulate air (HEPA) and charcoal filters. The CREF system is designed to provide a radiologically controlled environment from which the unit can be safely operated following a DBA. Two independent CREF subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CREF system automatically initiate action to start and direct flow through the control room outdoor air special filter trains and maintain the main control room envelope pressurized to minimize the consequences of radioactive material in the control room environment.

In the event of a LOCA signal (e.g., Reactor Vessel Water Level – Low, Level 2 or Drywell Pressure – High) or Main Control Room Ventilation Radiation Monitor – High signal, the CREF system is automatically started in the emergency pressurization mode. A portion of the control room envelope air is then recirculated through the charcoal filter, and sufficient outside air is drawn in through the two outside air intakes to keep the control room envelope slightly pressurized with respect to the outside atmosphere.

2.1.6 Standby Gas Treatment Filtration System

The standby gas treatment (SGT) filtration system plays a role in the AST dose calculations. However, input value changes have no impact on the TSs governing the SGT system. The secondary containment holds and dilutes the primary containment leakage into the reactor building (RB) prior to releasing to a high point via a vent stack, after first treating the releases through the SGT system filter train. The function of the SGT system is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a DBA are filtered and adsorbed prior to exhausting to the environment. The SGT system consists of two fully redundant subsystems, each with its own set of ductworks, dampers, charcoal filter train, and controls.

2.2 Description of Proposed Changes

2.2.1 Alternative Source Term Loss-of-Coolant Accident Radiological Analysis

In the LAR, the licensee proposed the following changes to the Nine Mile Point 2 AST LOCA radiological analysis assumptions, inputs, or methods:

Design Input Parameter	Current Licensing Value	Proposed Changed Value
Settling Velocity for Aerosol Deposition in the MSIV and system bypass (SB) Leakage Pathways.	AEB-98-03 3rd percentile for MSIV and SB leakage pathways.	3rd percentile settling velocity for SB leakage and 20-group probabilistic distribution of aerosol settling velocity for MSIV leakage based on AEB-98-03 and including RIS 2006-04 guidance.
Aerosol Deposition in Horizontal MSLs and Drywell and Wetwell System Piping.	Credited only between closed MSIVs and PCIVs. Not	Credited only between closed PCIVs for SB leakage and between the RPV nozzle and

Design Input Parameter	Current Licensing Value	Proposed Changed Value
	credited in MSL with one MSIV stuck open.	turbine stop valve (TSV) for MSIV leakage.
MSIV Leakage Rate.	24 standard cubic feet per hour (scfh) @ 40 psig (per MSL). 96 scfh @ 40 psig (total through all four MSLs).	100 scfh @ 40 psig (any one of the four MSLs). 200 scfh @ 40 psig (total through all four MSLs).
Drywell Spray Aerosol Iodine Removal Elemental Iodine Removal	Credited based on Standard Review Plan Section 6.5.2. Credited for 6.0 hours Credited for 3.157 hours	Credited with adjustments based on Standard Review Plan Section 6.5.2. Credited for 2.25 hours Credited for 2.4 hours
Holdup Time for Activity Releases via MSLs and SB Lines from Drywell	Credited based on plug flow	Not credited (includes well-mixed volumes)
SGT System Exhaust Rate	4,000 cubic feet per minute (cfm).	4,000 ± 10% cfm -4,400 cfm for airborne dose consequences and 3,600 cfm for RB shine dose.
CREF System Actuation Delay	50 seconds.	60 seconds.
Control Room Intake Flow Rates	2,750 cfm unfiltered (between 0 and 50 seconds). 2,750 cfm filtered (between 50 seconds and 20 minutes). 1,650 cfm filtered (between 20 minutes and 720 hours based on operator action to secure one ventilation train).	750 cfm unfiltered (between 0 and 60 seconds). 1,350 cfm filtered (between 60 seconds and 720 hours).
Core Inventory	GNF2 core inventory.	Based on modified fuel characteristics including increased core average exposure (CAVEX).
Dose Consequences Control Room Exclusion Area Boundary Low Population Zone	 1.65 rem TEDE 0.657 rem TEDE 0.769 rem TEDE	 2.27 rem TEDE 1.07 rem TEDE 0.91 rem TEDE

2.2.2 <u>Changes to the Delayed Drywell Leakage and Drywell Leakage Surveillance Requirements</u>

In the LAR, the licensee proposed to change SR 3.6.1.3.11 in TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," involving leakage rate for secondary containment. It is postulated that leakage from the primary containment through the PCIVs would bypass the RB and the SGT system filters, thereby resulting in an unfiltered ground-level release. This release pathway includes leakage through the MSIVs, as well as combined leakage through the piping of other systems such as the main steam drains, reactor water cleanup system, drywell floor and equipment drains and vents, post-accident sampling system, instrument air and nitrogen supply system, and primary containment purge system. The existing SR 3.6.1.3.11 divides these bypass pathways into the following three groups, excluding the MSLs:

- bypass from the drywell
- bypass from the wetwell (suppression pool)
- bypass from the drywell

SR 3.6.1.3.11 provides the frequency of verification and the maximum allowed leakage rate for the drywell and the suppression pool bypass paths. SR 3.6.1.3.12 provides the frequency and the maximum allowed leakage rate for each MSIV. TS 5.5.12.a.1 states that the measured leakage of MSIVs is excluded from the combined leakage rate L_a. TS 5.5.12.c defines L_a as the maximum allowable primary containment leakage rate at P_a and that it shall be 1.1 percent of primary containment air weight per day. TS 5.5.12.b defines P_a as peak calculated containment internal pressure for the design-basis loss of coolant accident and that it is equivalent to 39.75 pounds per square inch gauge (psig).

In the LAR, the licensee proposed the following changes to SR 3.6.1.3.11:

Existing SR 3.6.1.3.11	**Proposed SR 3.6.1.3.11**
SR 3.6.1.3.11 Verify the leakage rate for the secondary containment bypass leakage when pressurized to ≥ 40 psig is: a. Bypass (Drywell): ≤ 8.74 SCFH; and b. Bypass (Suppression Chamber): ≤1.67 SCFH; and c. Bypass (Drywell with delays): ≤ 28.17 SCFH	SR 3.6.1.3.11 Verify the leakage rate for the secondary containment bypass leakage when pressurized to ≥ 40 psig is: a. Bypass (Drywell): ≤ 36.88 SCFH; and b. Bypass (Suppression Chamber): ≤ 1.66 SCFH

2.2.3 <u>Change to the Allowable Main Steam Isolation Valve Leakage Rate Surveillance Requirement</u>

In the LAR, the licensee proposed to change SR 3.6.1.3.12 involving MSIV leakage rate. The current leakage rate limit of less than or equal to 24 scfh when tested at greater than or equal to 40 psig for each MSIV would be revised to allow a leakage rate of less than or equal to 50 scfh when tested at greater than or equal to 40 psig for each MSIV. The licensee stated that the changes to the leakage rate limits are based on a revised radiological analysis of the design-basis LOCA in accordance with the AST methodology and revised radiological analysis for EQ and vital area access, which are based on Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor Sites” (ADAMS Accession

No. ML021720780), source term methodology. The licensee's proposed changes affect the inputs and elements of the methodology for the AST LOCA calculation.

Existing SR 3.6.1.3.12	Proposed SR 3.6.1.3.12
SR 3.6.1.3.12 Verify leakage rate through each MSIV is ≤ 24 scfh when tested at ≥ 40 psig.	SR 3.6.1.3.12 Verify leakage rate through each MSIV is ≤ 50 scfh when tested at ≥ 40 psig.

2.3 Description of Regulatory Requirements and Guidance

2.3.1 Alternative Source Term Loss-of-Coolant Accident Radiological Analysis Requirements

Section 50.67 of 10 CFR states, in part:

> (2) The NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:
>
> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 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.

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 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 [0.05 Sv] 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 [10 CFR Part 50] who apply on or after January 10, 1997, … or holders of operating licenses using an alternative source term under [10 CFR] 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 [10 CFR] 50.2 for the duration of the accident.

Guidance

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), 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 AST (also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

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," dated July 2000 (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of AST amendment requests. SRP Section 15.0.1 states that the staff reviewer should evaluate the proposed change against the guidance in RG 1.183. RG 1.183 also states, in part, that an acceptable model for the reduction of airborne radioactivity in containment is provided in SRP Section 6.5.2, Revision 4, "Containment Spray as a Fission Product Cleanup System," dated March 2007 (ADAMS Accession No. ML070190178).

Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), provides guidance to ensure that the appropriate level of technical detail is considered in AST analyses and included in AST submittals.

Previous Approvals and Current Licensing Bases

The current design-basis LOCA radiological analysis for Nine Mile Point 2 was approved by the NRC staff in Amendment No. 125, dated May 29, 2008 (ADAMS Accession No. ML081230439). This amendment used an AST methodology for analyzing the radiological consequences of the DBAs using the regulatory guidance in RG 1.183.

The NRC staff also considered relevant information in the Updated Final Safety Analysis Report (UFSAR), which describes the DBAs and evaluation of their radiological consequences for Nine Mile Point 2.

2.3.2 Boron Precipitation

Implementation of AST LOCA radiological analysis at Nine Mile Point 2 requires use of the SLCS to control the pH level in the suppression pool during mitigation of AST LOCA. The safety concern is the possibility of boron precipitation in the core and potential flow blockage by the precipitates at Nine Mile Point 2 due to the initiation of SLCS.

Requirements

GDC 35, "Emergency core cooling," which states, in part, that a system to provide abundant emergency core cooling shall be provided to transfer heat from the reactor core following any loss of coolant at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented.

Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of 10 CFR, 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," states that calculated changes in core geometry shall be such that the core remains amenable to cooling.

2.3.3 Environmental Qualification

A change to increase MSIV leakage rate could challenge the environmental qualification of electrical equipment.

Requirements

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 states that the time-dependent temperature and pressure at the location of the electrical equipment important to safety must be established for the most severe DBA during or following which this equipment is required to remain functional. Section 50.49(e)(2) of 10 CFR states that humidity during DBAs must be considered. Section 50.49(e)(4) of 10 CFR states 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 nonsafety-related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety-related equipment.

2.3.4 Technical Specifications

The DBA analysis assumes that isolation of the primary containment is complete and leakage terminated, except for the maximum allowable leakage, L_a, prior to fuel damage. SR 3.6.1.3.11 for secondary containment bypass leakage paths and SR 3.6.1.3.12 for MSIVs ensure leakage is within the limits assumed in the accident analyses. This provides assurance that the assumptions in the radiological evaluations are met.

Requirements

Section 50.36(b) of 10 CFR requires, in part, that TSs be derived from the analyses and evaluation included in the safety analysis report.

Section 50.36(c)(iii)(3), "Surveillance requirements," of 10 CFR states, in part, that TSs shall include the "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."

3.0 TECHNICAL EVALUATION

3.1 Alternative Source Term Loss-of-Coolant Accident Radiological Analysis

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed changes and did independent calculations to confirm the conservatism of the licensee's analyses. However, unless otherwise noted, the findings of this safety evaluation (SE) are based on the descriptions of the analyses and other supporting information submitted by the licensee. The calculation (H21C-106, Revision 4) that the NRC staff reviewed was attached to the licensee's letter dated May 14, 2020.[1] The staff also considered relevant information in the UFSAR and the Nine Mile Point 2 TSs.

In the LAR, the licensee stated that the revised analysis of the LOCA radiological consequences performed to support the proposed license amendment is, in part, based on guidance provided in RG 1.183 and that changes to the radiological LOCA calculation's methodology and inputs are provided in Table 1 of Attachment 1 of the LAR, and as supplemented by letter dated May 14, 2020. The NRC staff, using the RG 1.183 guidance, evaluated the proposed changes.

3.1.1 Accident Source Term and Core Inventory

The licensee proposed to revise core inventories based upon modified fuel characteristics and increased cored average exposures used in the LOCA radiological calculation. The NRC staff used RG 1.183 guidance to evaluate the proposed change. RG 1.183, Regulatory Position (RP) 3.1, "Fission Product Inventory," states, in part:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty…. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values…. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 … or ORIGEN-ARP [Appendix F7.A in NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluation," dated May 2000].

The revised inventory of fission products in the reactor core and available for release to the containment is based on the maximum full power operation with a core thermal power of 4,067 megawatts thermal (MWt) (i.e., 102 percent of 3,988 MWt nominal power). Core design parameters assumed an increased core average exposure up to 37 gigawatt-days per metric ton of uranium (GWD/MTU) at 3.7 weight percent (wt.%) U-235 and 41 GWD/MTU from 4 wt.% to 4.5 wt.% U-235. The licensee also used an "equilibrium core inventory" (as stated on

[1] The licensee's initial request in the letter dated May 31, 2019, included two MSIV leak rate cases (i.e., 400 scfh and 200 scfh) with a calculation supporting both cases. In its supplemental letter dated May 14, 2020, the licensee revised its request and the calculation supporting the request to include only the 200 scfh case with corresponding changes to the calculation supporting the revised request.

page 31 of Calculation H21C-106, Revision 4) and calculated the inventory using ORIGEN-ARP.

Because the proposed inventories were derived from the ORIGEN-ARP code and based upon (1) bounding fuel inventories, (2) bounding fuel enrichments, (3) the Nine Mile Point 2 rated thermal power (with ECCS) uncertainty, and (4) equilibrium core inventory values, the NRC staff finds that core inventory used for the design-basis LOCA radiological analysis is consistent with RG 1.183, RP 3.1 and, therefore, acceptable for use in the DBA LOCA radiological analysis.

3.1.2 Control Room Intake Flow and Actuation Delay

The licensee proposed to revise the assumed control room intake flow and the control room filter actuation delay in the LOCA radiological calculations. The NRC staff used the guidance in RG 1.183 to evaluate the proposed change. RG 1.183, RP 5.1.3, "Assignment of Numeric Input Values," states, in part, that "The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative dose."

The control room operator dose is sensitive to control room ventilation flow rate and filter actuation timing. Per the LAR (see Attachment 1, page 14), a sensitivity analysis was performed to determine the appropriate and conservative flow rate and filter timing to use in the revised LOCA dose analysis. The results of this sensitivity analysis concluded that the parameters in Attachment 1, Table 3 of the LAR result in the highest control room doses. Therefore, the revised control room intake flow rates from Attachment 1, Table 3 and the assumed filtration actuation delay of 60 seconds were used in the revised LOCA radiological analysis. Based upon the results of the licensee's sensitivity analysis and the statements by the licensee that the assumed values yield the highest control room dose, the NRC staff finds that the revised values are consistent with RG 1.183, RP 5.1.3 and are, therefore, acceptable.

3.1.3 Standby Gas Treatment System Exhaust Rate

The licensee proposed to revise the assumed SGT exhaust flow in the LOCA radiological calculations. The NRC staff used RG 1.183 to evaluate the proposed change. In addition to stating that inputs should be selected to determine a conservative dose, RG 1.183, RP 5.1.3 states, in part, that:

> A single value may not be applicable for a parameter for the duration of the event.… For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications…. If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used.

Attachment 1, page 13 of the LAR states, in part, that:

> The SGT System exhaust flow rate varies from 3,600 cfm to 4,400 cfm, consistent with TS 5.5.7…. An SGT System exhaust flow rate of 4,400 cfm (10% higher than the CLB [current licensing basis] value) is used in the LOCA revised dose analysis because it maximizes the CR [control room] and offsite doses.

> This value of 4,400 cfm becomes non-conservative for the RB shine dose because it removes a larger amount of activity from the secondary containment volume leaving a lesser amount to contribute to the RB shine dose to CR personnel. Therefore, a value of 3,600 cfm (10% lower than the CLB value) is used in the RB shine analysis to maximize the post-LOCA activity confined above the RB operating floor, which conservatively increases the RB shine dose to the CR operators.

Based upon the licensee's statements that (1) it considered the range of SGT exhaust flow values (i.e., 3,600 cfm to 4,440 cfm consistent with those in TS 5.5.7) to maximize the postulated LOCA doses, (2) the SGT system flow rate of 4,400 cfm maximizes the control room offsite doses (due to containment leakage), and (3) the SGT system flow rate of 3,600 cfm maximizes the post-LOCA activity above the RB operating floor to increase the RB shine dose to the control room operators, the NRC staff finds that the revised assumptions for the assumed SGT exhaust flow rate are consistent with RG 1.183, RP 5.1.3 and are, therefore, acceptable.

In the TSs, the SGT system is addressed by TS 3.6.4.3, "Standby Gas Treatment (SGT) System," for operability and TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," for testing. In the CLB analysis, the licensee used 4,000 cfm as the system exhaust rate through the SGT system filters. In the revised analysis, the licensee used two different exhaust rates—3,600 cfm and 4,400 cfm. The licensee stated that the use of a higher rate of 4,400 cfm in the revised analysis will maximize the CR and offsite doses and the use of a lower flow rate will maximize the RB building shine dose to the control room operators.

The acceptance value for SGT system flow rate in TS 5.5.7 is in the range of 3,600 cfm and 4,400 cfm. Therefore, the method of calculating the doses as described will maximize the contributions to the CR operators and thus the NRC staff finds it acceptable. Further, the use of 99 percent filter efficiency for the HEPA and charcoal filters is consistent with the acceptance test values in TS 5.5.7.

The NRC staff finds that the inputs to the SGT system operation are acceptable, as they are consistent with TS 5.5.7. The separate system exhaust rates are acceptable, as the calculation and the methods used in the calculation have the ability to evaluate the doses due to incoming flow and shine separately with no interdependence.

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

In Attachment 1, page 5 of the LAR, the licensee states:

> The 20-group probabilistic distribution methodology has been previously approved at Clinton…, Limerick…, and LaSalle [ADAMS Accession Nos. ML052570461, ML062210214, and ML101750625, respectively].

The NRC staff notes that the cited precedents included a ruptured MSL to maximize the dose consequences from MSIV leakage. Appendix A of AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998 (ADAMS Accession No. ML011230531), 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 Attachment 1, page 8 of the LAR, which states:

> All MSLs in the MSIV leakage release pathways are seismically designed and supported to withstand the Safe Shutdown Earthquake (SSE) and thereby comply with RG 1.183, Appendix A, Section 6.5 requirement. The recirculation line break is the limiting event for fuel failure. It is not credible to assume two initiating limiting events, a recirculation line break and a break on the main steam line in a single design basis accident.
>
> All four MSL headers are Seismic I and QA Cat [Quality Assurance Category] 1 from the RPV nozzle to seismic boundary break at the TSV [turbine stop valve]; therefore, they are qualified to withstand the SSE, and they comply with the RG 1.183, Appendix A, Section 6.5 requirement to be credited for aerosol deposition. Therefore, the MSIV leakage pathway boundary is extended up to the TSV.

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 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.

The NRC staff issued request for additional information (RAI)-4 on February 14, 2020 (ADAMS Accession No. ML20045E358), 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 May 14, 2020, the licensee responded by first stating that its analysis submitted in the LAR conservatively only modeled MSLs “A” and “D,” which are the two shortest MSLs, and modeled 100 scfh MSIV leakage through each line. 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 in response to RAI-4. The licensee stated:

> The MSLs were modelled with three nodes; one from the reactor pressure vessel to the inboard MSIV, one for the volume between the MSIVs, and one for the volume between the outboard MSIV and the turbine stop valves. It was assumed that MSL “A” has a line break inside containment and a failed inboard MSIV. As

> a result, there was no credit taken for holdup (volume set to arbitrary small value in RADTRAD for this compartment) or aerosol iodine deposition in the line segment from the RPV nozzle to the inboard MSIV for the broken MSL. The flow in each of the four main steam lines was assumed to be 50 scfh in accordance with the proposed Technical Specification Surveillance Requirement 3.6.1.3.12. No other changes were made to the model or methodology described in the main body of H21C-106, Revision 4.

Additionally, the licensee provided a sensitivity analysis as part of the response to RAI-4. The NRC staff did not review or consider the sensitivity analysis provided in response to RAI-4.

3.1.5 Aerosol Settling Velocity for Aerosol Deposition in the MSIV Leakage Pathways Using a "20-group"

In the LAR, the licensee discussed it's review of NRC AST SEs previously issued for Nine Mile Point 2 [and other plants] and how these SEs discussed the staff's concern with how much deposition is assumed in the LOCA MSIV leakage pathways when using the AEB-98-03 model.

In the NRC staff's SE for Amendment No. 125, which approved the licensee's full implementation of the AST methodology for Nine Mile Point 2, the NRC staff indicated that it had concerns regarding the use of AEB-98-03. At that time, the NRC staff based its approval of the LAR, in part, upon additional conservatism in the deposition model used. Specifically, the SE states, in part:

> However, for additional conservatism, and to address [NRC staff] concerns historically documented by the NRC staff, the licensee used [1/2 of] the 3rd percentile settling velocity of 0.000066 m/sec [meters per second]. The NRC staff agrees that this [1/2 of the] 3rd percentile settling velocity value is sufficiently conservative to reflect the effectiveness of drywell spray activity removal in containment upstream of this pathway.

The NRC staff notes that in Nine Mile Point 2 Calculation No. H21C-106, "Unit 2 LOCA w/LOOP, AST Methodology" (ADAMS Accession No. ML071580354), previously transmitted to the NRC, page C2 indicates that one-half of the third percentile value is equivalent to the settling velocity of 0.000066 m/sec).

In Attachment 1, page 5 of the LAR, the licensee stated:

> The revised 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…, Limerick…, and LaSalle…. The same settling velocity probability distribution function shown in Equation 5 of AEB-98-03 is used to conservatively calculate aerosol settling velocity….

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" (ADAMS Accession No. ML063480542), provides details on how sprays impact aerosols. NUREG/CR-5966 indicates that the sprays shift the sizes of aerosols in the containment toward 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 are determined using, in part, Equation 5 of AEB-98-03. Equation 5 of AEB-98-03 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.

As discussed in the SE for Amendment No. 125, 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 the 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.

From an examination of the submitted information, the licensee considered 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 and, therefore, making subsequent removal by deposition in downstream piping more challenging.

In RAI-5 dated February 14, 2020, the NRC staff asked the licensee to provide technical information to describe how the gravitational settling credited in the MSLs using the 20-group method considers the changing aerosol characteristics (i.e., aerosol size and density distributions) due to the sprays as these aerosols move through the MSLs, and why the results of the 20-group method when crediting sprays are valid for Nine Mile Point 2.

Licensee's Sensitivity Study

In response to RAI-5, the licensee stated that a simplified model was developed using first principles as identified in NUREG/CR-5966. The ordinary differential equation shown on page 1 of NUREG/CR-5966 was solved to provide an analytical solution of the suspended aerosol mass in the drywell. The spray removal rate in the simplified model is the same as that identified in H21C-106, Section 2.1.3, and RG 1.183, Appendix A, Section 3.3. Since sprays will remove aerosols at different rates depending on their particle size, the spray removal rate is adjusted by collection efficiency variation as provided in Figure 19 of NUREG/CR-5966. The suspended aerosol mass was solved from the beginning of the accident through the termination of the drywell sprays at 2.25 hours for 20 distinct particle size groups. The mass of particles in each group is defined by the probability distribution associated with the source distribution.

The licensee assumed the size distribution of the particles released from the fuel was log-normal with 2-micron aerodynamic mass median diameter (AMMD) (0.473-micron geometric mean diameter) with a geometric standard deviation of 2. The aerosol mass was calculated for each group independently with no consideration of particles interacting with one another, so agglomeration was not considered. Table RAI-5.a, "Drywell Particle Size Distributions," summarizes the results of the 20-group particle size distribution in the drywell. Figure RAI-5.a, "Time-Dependent Aerosol Particle Size Distribution," shows the time-dependent nature of the aerosol particle size distribution and the effect of the drywell spray in reducing the size of the particles in the licensee's proposed model.

The particle size and settling velocity distributions were then used to recalculate the aerosol removal rate using the equations provided in Section 2.3.1.1 of H21C-106, Revision 4. The resulting aerosol removal factors are summarized in Table RAI-5d, "Steam Line and Condenser Aerosol Removal Factors." The aerosol removal factors, including spray, combined with the nodalization adjustments described below, are represented by the base sensitivity case ("base case") row in Table RAI-5e, "Sensitivity Study Results." The sensitivity analyses use the Nine Mile Point 2 RADTRAD model inputs.

A total of seven sensitivity cases were performed by varying the base case. The base case is essentially the H21C-106, Revision 4 model, including the MSL nodalization (increased number of holdup and deposition volumes), MSL break of the "A" inboard line, and flow rate distribution adjustments described in Attachment 13.19 (to address RAI-4 above), as well as the revised aerosol removal factors described above. As Table RAI-5e indicates, the seven sensitivity cases are various combinations of the three sensitivities described above (breathing rate, MSIV impaction, and condenser holdup). The sensitivity case results are summarized in Table RAI-5e.

The sensitivity analysis modified the nodalization of the MSL to overcome limitations of the RADTRAD code. The H21C-106, Revision 4, MSL nodalization described in the main body was modified to separately model each of the four MSLs as shown in Figure RAI-5b. As a result, each sensitivity case includes four RADTRAD models, one for each line with three well-mixed nodes per line. In the sensitivity analysis, MSL "A" is assumed to be broken, and the MSIV leakage is assumed to be equally distributed among all four MSLs. This modeling is consistent with the nodalization used in the H21C-106, Revision 4, Attachment 13.19, sensitivity cases created as part of the response to RAI-4, but not the proposed analysis of record used to show compliance with 10 CFR 50.67. The data used to calculate the steam line and condenser aerosol removal rates are provided in Tables RAI-5b and 5c and are consistent with H21C-106, Revision 4.

The NRC staff observed that the basis for the 2-micron AMMD particle size and the methodology for the 20-group particle size distribution were not fully described in the licensee's response to RAI-5 and, therefore, were not reviewed by the NRC staff. The NRC staff notes that neither the 2-micron AMMD particle size nor the 20-group particle size distribution method were described in the licensee's CLB for calculating design-basis LOCA doses.

The licensee assumed a 2-micron AMMD and geometric standard deviation of 2.0 particle size in the 20-group method to recalculate the aerosol removal rates in the licensee's sensitivity analysis. The NRC staff did not base its evaluation of this amendment on 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.

<u>NRC Staff's Evaluation of Licensee's Sensitivity Analysis</u>

In response to RAI-5, the licensee stated that a sensitivity analysis was performed to evaluate the impact of sprays on the aerosol settling velocity and to identify other inputs with well-defined uncertainty or conservatism that could be used to offset the uncertainty associated with the current aerosol deposition model. Based on the sensitivity analysis results, the licensee asserted that conservatism associated with modeling the total MSIV leakage as split between two lines, as opposed to split evenly among all four lines, with an assumed break in an MSL (to address RAI-4), approximately offsets the uncertainty introduced by the drywell spray effects on the aerosol deposition model. Other conservatisms explicitly evaluated in the sensitivity analysis are discussed below.

The licensee performed a total of seven sensitivity cases from various combinations of breathing rate, MSIV impaction, and condenser holdup/aerosol deposition by varying the base case. The sensitivity case results are summarized in Table RAI-5e. The licensee stated that:

> As expected, the base case indicates the conservative modelling of the drywell spray impact on the aerosol removal in the main steam lines without adjusting any other inherent conservatisms in the RADTRAD inputs results in increased doses. Because the MSIV leakage portion of the Control Room dose is only 0.40 rem in the base analysis (H21C-106, Revision 4, Attachment 13.19), the increase (~0.26 rem) due to the revised aerosol removal rates does not increase the Control Room dose above the 5 rem limit. Rather, as the results in Table RAI-5e indicate, the base case doses are within 2% of the previously submitted results (shown as H21C-106 Main Body).

The NRC staff notes that the licensee's base case was produced for the purpose of conducting a sensitivity analysis and does not replace the proposed accident analysis of record, which is provided in the revised design-basis LOCA radiological analysis Calculation H21C-106, Revision 4, and was revised in response to RAI-6. The accident analysis of record indicates that dose consequences comply with all the applicable dose acceptance criteria.

The licensee asserted that the six bulleted items listed below could be used to address reduced aerosol removal rates due to drywell sprays. The NRC staff's evaluation is provided under each of these asserted conservatisms in the AST LOCA model.

- Credit full drywell spray lambdas (not included in the licensee's evaluation)

  As stated by the licensee, credit for full drywell spray lambdas was not included in the licensee's evaluation. Because an evaluation and technical basis were not included in the licensee's sensitivity study, the NRC staff did not review the licensee's assertion that credit for full drywell spray lambdas is a conservatism in the licensee's AST LOCA model and reasonable assurance that the acceptance criterion will not be exceeded. Therefore, the NRC staff does not consider credit for full drywell spray lambdas to be a conservatism in the AST LOCA model.

- Credit for plateout and deposition in drywell (not included the licensee's evaluation)

  As stated by the licensee, credit for plateout and deposition in the drywell was not included in the licensee's evaluation. Because an evaluation and technical basis was not included in the licensee's sensitivity study, the NRC staff did not review the licensee's assertion that the proposed model's lack of credit for plateout and deposition in the drywell is a conservatism in the AST LOCA model. Sprays, plateout, and deposition in the drywell impact the aerosol distribution removal in the MSLs. However, the LAR analysis of record does not consider the impact of plateout or settling in the drywell on the credited setting in the MSL. Therefore, the NRC staff does not consider credit for plateout and deposition in the drywell to be a conservatism in the AST LOCA model.

- Inclusion of all four MSLs for holdup and deposition with flow split evenly among all four lines

  Modeling all four MSLs for some holdup and deposition with the flow split evenly among all four lines is consistent with the proposed Nine Mile Point 2 TSs. However, there is no sensitivity case or results provided that assess only the impact of modeling four lines as opposed to two lines. Also, the sensitivity cases provided, which included four steam lines, also modeled up to three nodes for deposition as opposed to the proposed analysis of record, which only modeled up to two nodes of deposition in each steam line. Thus, the impact of modeling two lines rather than four lines is unknown. The NRC staff expects that modeling all four lines could be a conservatism but is unable to acknowledge the impact of modeling all four lines and whether it is a conservatism.

- Operator breathing rate

  The licensee referenced breathing rate data from Table 6-17 of U.S. Environmental Protection Agency (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 provides a method acceptable to the NRC staff for demonstrating compliance with 10 CFR 50.67 and uses a constant value of 3.4E-04 cubic meters per second (m^3/s) for the duration of the control room dose consequence analysis. In Table RAI-5e, the licensee used the RG 1.183 recommended breathing rate for the first 2 hours, followed by reduced breathing rates that the licensee asserted were taken from the EPA handbook (3.28E-04 m^3/s from 2 to 12 hours and 3.06E-04 m^3/s from 12 hours to 30 days). The licensee stated that it considered the 95 percent data values from the EPA handbook as light intensity work typical of a control room operator. As a result, the observed control room dose was reduced when compared to the base sensitivity case.

  From the NRC staff's examination of the sensitivity cases (S1, S4, S6, and S7) when compared to the base sensitivity case (S0) in Table RAI-5e, consideration of a more realistic control room breathing rate is observed to show a reduction of the control room dose. 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 is not considered justified for determining 10 CFR 50.67 design-basis radiation exposures from "access to and occupancy of" the control room under accident conditions where control room operators would be expected to be at a higher level of stress and increased activities. Therefore, use of a more realistic control room breathing rate would not be considered in the NRC's design-basis determination of its acceptability. In addition, the consideration of a more realistic control room breathing rate would result in an NRC RG assumption change in the dose calculational methodology for the MSIV leakage dose consequence analysis. Therefore, the NRC staff does not consider credit for a more realistic control room breathing rate to be a conservatism in the AST LOCA model.

- Aerosol impaction on the first closed MSIV

  In Table RAI-5e, the licensee considered credit for aerosol impaction on the first-closed MSIV in its RADTRAD model. As a result, the observed control room, exclusion area boundary (EAB), and low population zone (LPZ) doses were reduced when compared to the base sensitivity case.

  From the NRC staff's examination of the sensitivity cases (S2, S5, S6, and S7) when compared to the base sensitivity case (S0) in Table RAI-5e, consideration of MSIV impaction is observed to show a reduction of the control room, EAB, and LPZ doses. The licensee referenced the Nine Mile Point 1 AST LOCA licensing basis described in Calculation H21C092, "U1 LOCA w/LOOP, AST Methodology" (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. The licensee stated that for Nine Mile Point 1 this impaction results in a decontamination factor (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 asserted that this approach was previously approved for Nine Mile Point 1 by Amendment No. 194, dated December 19, 2007 (ADAMS Accession No. ML073230597), and is reasonable, given that the aerosol settling rates calculated in this sensitivity analysis are conservative and lower than those used in the cited analysis.

The NRC staff acknowledges that the following assumption 7 from the Nine Mile Point 1 AST LOCA Calculation H21C092 was included in its MSIV leakage dose consequence analysis:

> 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….

The NRC staff's SE associated with Nine Mile Point 1 Amendment No. 194 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 [Nine Mile Point 1]. 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 the licensee credits for removal by sedimentation…, 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 the licensee to be acceptable, as modeled for [Nine Mile Point 1].

The NRC staff notes that the above excerpt from the staff's Nine Mile Point 1 Amendment No. 194 SE clearly states that the staff does not generally endorse taking credit for impaction when modeling removal of particulates in main steam lines following a LOCA. The NRC staff's SE concluded that notwithstanding the issue of credit for impaction, the overall iodine removal credited as modeled for Nine Mile Point 1 was acceptable. Under 10 CFR 50.40, the NRC staff makes determinations based on the collective circumstances of an application. Absent all those circumstances, it could reach a different determination. Therefore, this conclusion should not be interpreted as an NRC staff acceptance of credit for impaction when modeling removal of particulates in main steam lines following a LOCA. Thus the NRC staff does not consider credit for MSIV impaction to be a conservatism in the AST LOCA model.

- Condenser holdup and deposition

  The licensee stated that a further conservatism that is not currently modeled in H21C-106 is the holdup and aerosol deposition provided by the condenser. The licensee asserted 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 neglects 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 County Station and Limerick Generating Station). The licensee 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. 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 licensee stated that the data used to calculate the steam line and condenser aerosol removal rates provided in Tables RAI-5b and 5c are consistent with Calculation H21C-106, Revision 4.

  In Table RAI-5e, the licensee considered condenser credit in its RADTRAD model. As a result, the observed control room, EAB, and LPZ doses are effectively reduced when compared to the base sensitivity case. The licensee stated that condenser credit has the capability to ensure post-LOCA releases remain well within the 10 CFR 50.67 limits.

  From the NRC staff's examination of the sensitivity cases (S3, S4, S5, and S6) compared to the base sensitivity case (S0) in Table RAI-3e, consideration of condenser credit is observed to show a reduction of the control room, EAB, and LPZ doses. The licensee's estimate of the control room doses when crediting the condenser are about 28 percent less than for the base sensitivity case, which does not credit the condenser.

  The licensee's sensitivity results to consider condenser credit were observed to be effective in reducing the dose consequences from MSIV leakage due to the condenser's mitigation properties. The NRC staff notes that while the guidance provided in RG 1.183 for design-basis LOCA radiological analysis states that the structures, systems, and components (SSCs) credited with creating a pathway to the condenser shall be able to withstand an SSE, it is reasonable to consider the probability of the existence of a pathway to the condenser to offset the uncertainties in the calculation of the dose consequences of MSIV leakage. The NRC staff's consideration of risk and engineering insights is discussed in Section 3.10 of this SE.

RG 1.183, Appendix A, describes assumptions for evaluating the radiological consequences of a LOCA. Section 6 of Appendix A describes assumptions on MSIVs in BWRs. Specifically, 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.

The licensee also stated that there are other significant conservatisms associated with the AST LOCA model. Specifically, the licensee stated that control room atmospheric dispersion (χ/Q) factors (values) have readily defined uncertainty distributions and, if incorporated, would demonstrate that there is a substantial amount of margin in the input parameters. The licensee further stated that for simplicity, the distribution of potential values for such input parameters were not evaluated in the sensitivity study. The NRC staff notes that the use of χ/Q values in design-basis dose consequence analyses is a well-established practice and should not be included in sensitivity analyses. Atmospheric dispersion values 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 χ/Q values used in design-basis dose consequence analyses should not be included in sensitivity analyses.

3.1.6 <u>Elemental Iodine Removal Rate and Deposition in Main Steam Line Piping</u>

In its LAR, the licensee proposed to change the elemental iodine removal credited in the steam lines between the RPV nozzle and the turbine stop valve. RG 1.183, Appendix A, RP 6.3, states, in part, that a reduction in the amount of radioactivity upstream of the outboard MSIVs may be credited, but the amount of reduction is evaluated on an individual case basis. RG 1.183, Appendix A, RP 6.5, states, in part, that a reduction in the MSIV releases due to deposition in the main steam piping downstream of the MSIVs may be credited if the components and piping systems used are capable of performing their safety function during and following an SSE, and that the amount allowed will be evaluated on an individual case basis.

The NRC staff reviewed the proposed changes in credited elemental iodine removal in the MSL piping. The NRC staff identified technical and regulatory concerns regarding the proposed justifications for the proposed revised elemental iodine removal credit. To address these concerns, the NRC staff issued RAI-6 to the licensee on February 14, 2020.

By letter dated May 14, 2020, the licensee responded to RAI-6. To address this RAI, instead of using the J. E. Cline model to determine the MSL elemental iodine removal, the proposed revised calculation H21C-106, Revision 4, implements a constant 50 percent elemental iodine removal efficiency in the MSLs. The licensee asserted that this assumption is consistent with the CLB calculation, H21C-106, Revision 2, which was previously approved by the NRC.

The NRC staff compared the licensee's statement regarding the 50 percent elemental iodine removal (DF of 2) to the NRC staff's SE for Amendment No. 125. The SE for Amendment No. 125 states that for elemental iodine, the licensee assumed a DF of 2 (50 percent elemental iodine removal) in the bypass piping (see Section 3.2.1.2.5.2). Furthermore, the SE for Amendment No. 125 states that no credit is taken for deposition in the MSL with one MSIV stuck open (see Section 3.2.1.2.5). However, the proposed revised LOCA model provided with the response to RAI-6 credits elemental deposition in the MSL with one MSIV stuck open.

The NRC staff evaluated the impact of including credit for elemental deposition in the line with one MSIV stuck open and found that this proposed credit did not have a significant effect on the overall doses (using the proposed design-basis assumptions and calculational model for Nine Mile Point 2). Based upon (1) the licensee's statements that the proposed assumption is consistent with its CLB, and (2) the determination that there is no significant impact in the doses when crediting 50 percent elemental deposition removal in the line with the MSIV stuck open (for the proposed Nine Mile Point 2 design basis and calculational model), the NRC staff finds the modeling of the elemental iodine removal rate in the main steam piping to be effectively consistent with the CLB and, therefore, acceptable.

3.1.7 Drywell Spray – Aerosol and Elemental Iodine Removal

The CLB for crediting drywell spray is based on SRP Section 6.5.2. In the CLB, the iodine removal efficiency of 19.8 per hour is reduced by a factor of 10 at 2.017 hours. For elemental iodine, the removal efficiency is terminated at 3.157 hours, while the aerosol removal efficiency is assumed to continue until 6.0 hours.

In Attachment 1, page 11 of the LAR, the licensee stated that in the revised LOCA radiological analysis, the drywell spray removal of elemental iodine and aerosol is conservatively not credited after the respective DFs are reached. This change conservatively increases the drywell aerosol source term for MSIV leakage. In Table 1 of the LAR, the licensee continues to credit SRP Section 6.5.2 models with adjustments. The proposed aerosol iodine removal credit is credited for 2.25 hours, and the elemental iodine removal is credited for 2.4 hours.

The NRC staff reviewed the proposed changes in credited aerosol and elemental iodine removal credit using the guidance in RG 1.183 to evaluate the proposed changes. RG 1.183, Appendix A, RP 3.3, states, in part, that "Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP…."

Based upon the statement in the LAR that drywell spray continues to be credited based upon SRP Section 6.5.2 and the more conservative changes to not credit elemental and iodine removal after the respective DFs are reached, the NRC staff finds the proposed changes to be more conservative than the CLB use of SRP Section 6.5.2 and RG 1.183 and, therefore, acceptable.

3.1.8 Plug Flow Delays

The CLB LOCA radiological analysis includes credit for holdup of activity releases via the MSLs (based on MSIV leakage of 24 scfh per line) and system bypass (SB) lines from the drywell (feedwater, 14" containment purge, and reactor water cleanup). The delays credited in the CLB LOCA radiological analysis are as follows:

- 5.26 hours for the steam line with one MSIV failed open
- 7.11 hours for the steam line with both MSIVs closed
- 2.45 hours for the bypass from the drywell

The revised LOCA radiological analysis does not credit delay of the activity releases via the MSLs and the bypass from the drywell. Instead, the MSLs are modeled using well-mixed volumes. This proposed change is consistent with: (1) RG 1.183, Appendix A, RP 6.3, which states, "Generally, the model should be based on the assumption of well-mixed volumes…"; (2) the proposed changes to SR 3.6.1.3.11, which would remove the above delays; and (3) the revised LOCA radiological analysis and, therefore, is acceptable.

3.1.9 <u>Main Steam Isolation Valve and System Bypass Leakages</u>

The LAR would revise TS 3.6.1.3 by revising SR 3.6.1.3.12 for MSIV leakage rate. The current leakage rate limit of less than or equal to 24 scfh for each MSIV would be revised to allow a leakage rate of less than or equal to 50 scfh for each MSIV. The total allowable MSL leakage rate through all four steam lines would increase from 96 scfh to 200 scfh.

The NRC staff used RPs from Section 6, "Assumptions on Main Steam Isolation Valve Leakage in BWRs," in Appendix A of RG 1.183, to evaluate the proposed changes in MSIV leakage rates. RG 1.183, Appendix A, RPs 6 and 6.2, state, in part, that an assumption acceptable to the NRC staff for evaluating the consequences of MSIV leakage is that "All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable." The proposed LOCA radiological analysis that incorporates the proposed changes in MSIV leakage (see SR 3.6.1.3.12) is consistent with the maximum leak rate at which the MSIV would be required to be declared inoperable and, therefore, the proposed change is acceptable.

Attachment 1, page 19 of the LAR states:

> In addition, since the revised Loss of Coolant Accident (LOCA) analysis does not credit delays of activity releases via the bypass from the drywell (feedwater, 14" containment purge, and reactor water cleanup), SR 3.6.1.3.11 is revised by consolidating the total bypass from the drywell without accounting for delays. Small changes to the total drywell and wetwell bypass leakage rates in SR 3.6.1.3.11 are made to support the revised Alternative Source Term (AST) LOCA radiological analysis.

The letter transmitting the LAR states:

> The proposed change would revise TS Surveillance Requirement (SR) 3.6.1.3.11 to combine delayed drywell leakage from SR 3.6.1.3.11c with drywell leakage in SR 3.6.1.3.11a and delete SR 3.6.1.3.11c from the SR. The total drywell leakage in SR 3.6.1.3.11a and the wetwell leakage in SR 3.6.1.3.11b would also be revised to be consistent with the revised Alternative Source Term (AST) Loss-of-Coolant Accident (LOCA) radiological analysis.

The NRC staff used RG 1.183, Appendix A, RP 4.5, to evaluate the proposed changes in bypass leakage. RG 1.183, Appendix A, RP 4.5, states, in part, “Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications.” The proposed LOCA radiological analysis that incorporates the proposed changes in bypass leakage is consistent with RG 1.183, Appendix A, RP 4.5, and the proposed changes to SR 3.6.1.3.11; therefore, the proposed change is acceptable.

3.1.10 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, Revision 3, “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 staff requirements memorandum (SRM) to SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves” (ADAMS Accession No. ML19183A408), 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 to SECY-19-0036, the NRC’s efforts to advance the reactor safety program toward becoming a more modern and risk-informed regulator, and the NRC’s Principles of Good Regulation. Since the LAR is not a fully risk-informed submittal (with probabilistic risk information), the staff does not apply risk as the basis for approval of the LAR. 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 of reasonable assurance.

The licensee stated that aerosol holdup and deposition provided by the condenser are not modeled in H21C-106 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 NRC staff used engineering information such as operations and design knowledge, as well as risk information to complete the evaluation. The staff also leveraged more recent relevant operating experience such as that obtained from the Fukushima Daiichi accident and the August 23, 2011, magnitude 5.8 earthquake that impacted the North Anna Power Station. The staff’s independent assessment found that it is reasonable to conclude that the SSCs in the PCS would be available following an SSE and that the likelihood of the PCS being unavailable to serve as a volume for holdup and retention is very low.

The NRC staff’s independent 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 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 NRC 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 decision making, 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 the staff's finding based on its deterministic review.

3.1.11 Other Information Regarding the NRC Staff's Review

The LAR included calculation H21C-106, Revision 4. The NRC staff did not explicitly review and evaluate all of the details provided in H21C-106. For example, H21C-106, Revision 3, also provides calculations assuming a total MSIV leakage rate of 400 scfh (versus the 200 scfh requested in the LAR) and a discussion of conservatisms (pages 5-6). Because this additional information is not necessary to make a regulatory finding of compliance with 10 CFR 50.67 and GDC 19 for the proposed LAR changes, the NRC staff's review did not evaluate or make a regulatory finding regarding this additional information.

3.1.12 Dose Consequences for the Control Room and Offsite

In the Nine Mile Point 2 TSs, the CREF system is addressed by TS 3.7.2, "Control Room Emergency Filtration (CREF) System," for operability and TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," for testing. The acceptance value for the CREF system flow rate in TS 5.5.7 is between 2,025 cfm and 2,475 cfm.

The following changes were made in the revised AST analysis:

- The CLB assumes a delay of 50 seconds for filtration initiation. The delayed analysis conservatively increased the delay to 60 seconds.

- Control room habitability unfiltered in-leakage flow testing performed with tracer gas indicates negligible in-leakage. However, a highly conservative value 250 cfm is assumed for unfiltered in-leakage throughout the analyzed time of 720 hours (30 days).

In addition, based on a sensitivity analysis performed to determine the highest CR doses, the revised AST analysis used the following CREF system intake flows and timing:

- From 0 to 60 seconds, 750 cfm unfiltered intake flow of unknown origin with an additional 250 cfm of unfiltered in-leakage flow is conservatively assumed under the control room habitability testing. Between 0 to 60 seconds, the CREF system is getting initiated and the normal CR intakes are shutting down. Even though the source of the 750 cfm of unfiltered intake flow is not clearly defined, based on the sensitivity analysis performed to maximize the doses, the NRC staff finds that the input change is acceptable.

- From 60 seconds to 720 hours, 1,350 cfm of filtered intake flow and 675 cfm of filtered recirculation flow with an additional 250 cfm from control room habitability testing is assumed. The sum of the 1,350 cfm of filtered intake and 675 cfm of filtered recirculated flow is equivalent to 2,025 cfm, which is the lower limit of CREF flow in TS 5.5.7.

Based on the above, the NRC staff finds that the inputs to the CREF system operation are conservative and supported by TS 5.5.7.

The licensee proposed to revise the CLB LOCA radiological consequence analysis methods, assumptions, and inputs as described in the LAR and evaluated by the NRC staff above. The results of the revised analysis are provided in Table 1 of this SE and are compared to the 10 CFR 50.67 regulatory acceptance criteria.

Table 1
Nine Mile Point 2 LOCA Radiological Consequences
Expressed as TEDE (1) (rem)

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (rem) Receptor Location		
	Control Room	EAB (2)	LPZ (3)
Containment Leakage	0.468	0.312	0.364
ESF Leakage	0.348	0.185	0.179
MSIV Leakage	0.620	0.135	0.179
Reactor Building Shine	0.059	N/A	N/A
External Cloud Shine	0.073	N/A	N/A
CR Filter Shine	Negligible	N/A	N/A
Total Dose	2.27	1.07	0.91
Acceptance Criteria	5	25	25

(1) Total effective dose equivalent
(2) EAB maximum 2-hour dose
(3) LPZ 30-day dose at the outer boundary

The NRC staff reviewed the proposed revised Nine Mile Point 2 LOCA radiological analysis and focused its review on the proposed revised MSIV leakage model that used the guidance in RG 1.183 to show compliance with 10 CFR 50.67. RG 1.183, Appendix A, Section 6, states, in part, that “The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated consequences from the LOCA.” The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee’s methods, assumptions, and inputs. Based upon its review, the NRC staff concludes that the EAB, LPZ, and control room radiological doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and, therefore, are acceptable.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed license amendment at Nine Mile Point 2. The NRC staff finds the analysis methods and assumptions consistent with the applicable regulatory requirements and guidance. The NRC staff concludes with reasonable assurance, based in part on the risk insights to compensate for uncertainties in the evaluation of the dose consequences from the MSIV release pathway, that the licensee's estimates for the EAB, LPZ, and control room doses will comply with the cited acceptance criteria. The NRC staff further finds with reasonable assurance that Nine Mile Point 2, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of a design-basis LOCA.

3.2 Boron Precipitation

The provisions of 10 CFR 50.46 do not require SLCS, and SLCS is not used to satisfy the 10 CFR 50.46 acceptance criteria during a design-basis LOCA. However, implementation of AST LOCA radiological analysis at Nine Mile Point 2 requires use of SLCS to control the pH level in suppression pool during mitigation of AST LOCA.

Because SLCS is relied upon for radiological analysis of AST LOCA at Nine Mile Point 2, the objective of this review is to examine whether use of sodium pentaborate solution from SLCS following AST LOCA could result in boron precipitation and flow blockage in the core during the long-term cooling phase causing degraded core cooling.

The licensee proposes to use SLCS, 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 staff evaluated whether it is likely for boron injected from SLCS to precipitate in the core causing flow blockage and degrading core cooling. The staff believes that because the rates at which ECCS water is injected by core spray (CS) at the top of the core and by RHR 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 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 by letter dated June 29, 2018 (ADAMS Accession No. ML18078A061), are also applicable to potential issues related to boron precipitation if it occurs 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 (GEH) Transient Reactor Analysis Code (TRACG) code (NEDE-33005P-A, Revision 1, “TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,” dated February 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 above, the NRC staff finds that in the 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 SLCS injection at Nine Mile Point 2. As such, the staff further concludes that the proposed license amendment is acceptable because it will continue to satisfy 10 CFR 50.46(b)(4) and GDC 35 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 clad damage that could interfere with continued effective core cooling will be prevented, and that the core remains amenable to cooling.

The NRC staff concludes that SLCS injection during AST LOCA is not expected to result in boron precipitation. In the unlikely event of boron precipitation inside the core and fuel filter blockage, a single CS pump could provide adequate core cooling. Therefore, the NRC staff further concludes that the applicable regulations and requirements will continue to be met, adequate defense in depth will be maintained, and sufficient safety margins will be maintained.

3.3 Environmental Qualification

The licensee evaluated the impact of the increase in the MSIV leakage rate on compliance with 10 CFR 50.49. The licensee stated that the radiation source term basis for the EQ analyses is TID-14844, which is consistent with Nine Mile Point 2’s CLB. Only areas outside of secondary containment (SC) are potentially impacted by increased MSIV leakage rate.

The licensee stated that its evaluation of EQ impact outside SC included updating the EQ dose analyses with revised airborne doses resulting from increased MSIV leakage. The licensee revised total integrated doses (TID) for all EQ zones outside SC using updated post-LOCA doses and 60-year normal doses. The licensee compared the revised EQ zone TIDS to the current zone doses and the EQ zone classification threshold as follows:

- Mild: TID < 1.0E+3 rad
- Mild Except for Electronics (ME): 1.0E+3 < TID <1.0E+4 rad
- Harsh: TID > 1.0E+4 rad

The licensee's analyses and evaluation confirmed that no EQ zones transition from mild to ME and that no zones transition from ME to Harsh as a result of the license amendment. Based on no zones transitioning, the licensee concluded that there is no equipment that needs to be evaluated for inclusion in the EQ program.

For each EQ zone outside the SC that is currently classified as ME or Harsh based on the TID, the licensee identified and reviewed all environmental qualification document packages containing equipment in those zones for impact. The ME and Harsh areas outside SC that contain EQ equipment are zones in the SGT Building in proximity to the SGT filters, in the Control Building near the emergency ventilation filters, in the Auxiliary Service Building, and in the Screenwell Area. In the case of the zones in the SGT, the increased airborne dose from the change in MSIV leakage is negligible compared to the current TID in which the dose primarily comes from filter shine. Regarding the Control Building zones, the equipment in question is a temperature indication controller that is installed in several other EQ zones, including some inside SC. The TID associated with the zones inside SC where these controllers are installed bounds the updated TID for the Control Building zones. For the Auxiliary Service Building zones, the EQ equipment is a flow switch that is also installed in SC zones. The TID associated with the zones inside SC where these switches are installed bounds the updated TID for the Auxiliary Service Building zones. Similarly, for the Screenwell Area, the EQ equipment of interest (level switches), are also installed in zones inside containment, which have TIDs that bound the revised TID for the Screenwell Area. As such, equipment already included in the EQ program continues to be qualified for the radiological environment resulting from the proposed increased allowable MSIV leakage.

In the LAR, the licensee provided an evaluation of the radiological impact on the EQ of electrical equipment due to the proposed increased leakage rate of MSIVs. However, it did not provide an evaluation of the impact of the increased leakage rate on temperature, pressure, or humidity of electrical equipment in those zones of impact. In addition, the licensee did not address whether, considering the total dose expected (TID analysis of record), the change could result in electrical equipment currently classified as non-environmental qualified now being subject to the requirements of 10 CFR 50.49 (i.e., transition from a Mild area to Harsh). It was also unclear to the NRC staff whether the licensee considered the impact of the proposed change on nonsafety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. Accordingly, by letter dated October 23, 2019 (ADAMS Accession No. ML19296A186), the staff requested the licensee to provide additional information.

In its November 21, 2019, response to the NRC staff's RAI, the licensee stated that as documented in the Nine Mile Point 2 EQ program basis document (2EQPBD), for equipment located outside containment, harsh humidity, temperature, or pressure conditions will only exist in those areas affected by high energy line breaks, moderate energy line cracks, or DBAs. Outside containment, the LOCA only results in increased gamma and beta radiation doses from various contributions depending on its location in the plant (e.g., shine from containment, shine from recirculating fluids, leakage from containment including MSIV leakage, and shine from buildup on the SGT systems or Control Building filters). The licensee also stated that various post-LOCA leakage pathways from containment, including MSIV leakage, are released directly to the environment from appropriate release points. The releases are transported to various areas outside containment via atmospheric dispersion, rather than being released directly into plant areas outside containment as in the case of high energy line breaks. The release and transport through the atmosphere preclude any temperature, pressure, and humidity effects in EQ zones outside containment. The licensee further stated that the proposed change does not make any physical modification to the plant, but, rather, increases allowable leakage through an existing leakage path. As such, there is no change to the MSIV release pathway other than the allowable flow rate, and the only potential impact to the EQ program from the proposed change is to radiation in zones outside containment due to atmospheric dispersion.

In its November 21, 2019, response to the NRC staff's RAI, the licensee included Table 1, "EQ Zone TID Changes Due to MSIV Leakage Increases," which summarizes EQ zone TID changes due to the proposed increase to the MSIV leakage rate. Table 1 includes the EQ zone grouping, normal dose, current LOCA dose, updated LOCA dose, current TID, zone classification based on TID, and updated TID columns. The table includes 60-year normal doses through the period of extended operation.

The NRC staff reviewed the licensee's response, including Table 1, and confirmed that no EQ zones transition from Mild to ME or from ME to Harsh as a result of the proposed change. Furthermore, environmentally qualified electrical equipment in zones outside containment that would experience an increase in radiological dose has been qualified in areas of the plant that are exposed to significantly higher post-accident doses (e.g., inside containment). Therefore, the staff finds that the EQ of electrical equipment will not be adversely affected by the proposed increase in MSIV leakage. The staff also confirmed that no zones transition to a harsher classification and that electrical equipment currently in the Nine Mile Point 2 EQ program should remain qualified for the expected post-accident environment because of the proposed change. The staff also confirmed that there is no new equipment, including nonsafety-related equipment, that needs to be included in the Nine Mile Point 2 EQ program because of the proposed change.

Based on its review of the information in the LAR, as supplemented, the NRC staff finds that the EQ of electrical equipment will not be adversely impacted by the proposed change. Therefore, the staff concludes that the proposed change is acceptable with respect to EQ.

3.4 Technical Specifications

The licensee provided an evaluation of the proposed changes in Attachment 1 to the LAR. The licensee also provided a revised AST LOCA radiological analysis supporting the proposed changes in Enclosure A of the LAR. The staff reviewed the analysis and evaluation provided by the licensee.

Changes to the Delayed Drywell Leakage and Drywell Leakage Surveillance Requirements

The system bypass pathways are addressed in Sections 2.3.3 and 5.5.6 of the revised analysis. There are 25 bypass leakage pathways. In the treatment of releases, they are grouped together based on their release locations and the origin of the radioactive sources.

The surveillance of the bypass pathways is carried out by SR 3.6.1.3.11. The existing surveillance divided the verification of the leakage rate into three subgroups: SR 3.6.1.3.11.a verifies bypass from drywell (delays neglected), SR 3.6.1.3.11.b verifies bypass from the suppression pool, and SR 3.6.1.3.11.c verifies leakage from drywell (delays considered).

In Attachment 1, Section 4.2, "No Significant Hazards Consideration Analysis," the licensee addressed the proposed change to SR 3.6.1.3.11. The revised AST analysis does not credit delays of activity releases via the bypass from the drywell. Therefore, the licensee is proposing to delete SR 3.6.1.3.11.c and combine the affected leakage into SR 3.6.1.3.11.a. The NRC staff noted that the combined leakage value proposed for the revised SR 3.6.1.311.a is slightly different than the total of the current leakage values in SR 3.6.1.3.11.a and SR 3.6.1.3.11.c. In addition, the staff also noted that the leakage value for SR 3.6.1.3.11.b was also slightly different than the current TSs. The licensee stated in Section 4.2 that "Small changes to the total drywell and wetwell bypass leakage rates in SR 3.6.1.3.11 are made to support the revised Alternative Source Term (AST) LOCA radiological analysis." The difference in the values is very small and insignificant to make any impact on the results of the analysis and, therefore, the staff did not pursue any clarifications regarding the licensee's statement.

The NRC staff determined that the proposed changes to SR 3.6.1.3.11 would consolidate the drywell bypass leakage acceptance criteria into a single criterion. The staff determined that the consolidation of the acceptance criteria for drywell bypass leakage was justified by the licensee's analysis and evaluation provided in the LAR. Likewise, the staff determined that the slight reduction in suppression chamber bypass leakage acceptance criteria was justified by the licensee's analysis and evaluation provided in the LAR. The NRC staff concludes that the changes to SR 3.6.1.3.11 are acceptable because they will continue to be based on the analyses and evaluations in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34, as required by 10 CFR 50.36(b).

Change to the Allowable Main Steam Isolation Valve Leakage Rate Surveillance Requirement

The NRC staff determined that the proposed increase in MSIV leakage acceptance criteria will increase the dose consequences of MSIV leakage; however, the staff has confirmed that the regulatory requirements related to dose will continue to be met. The NRC staff determined that the changes to SR 3.6.1.3.12 are acceptable because they will continue to be based on the analyses and evaluations in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34, as required by 10 CFR 50.36(b).

The NRC staff also determined that the SRs, as modified by the proposed changes, will continue 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 as required by 10 CFR 50.36(c)(3). Therefore, the staff concludes that the proposed changes are acceptable.

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 September 2, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components 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 (84 FR 47547). 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 staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: October 20, 2020

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 2 - ISSUANCE OF AMENDMENT NO. 182 TO CHANGE ALLOWABLE MAIN STEAM ISOLATION VALVE LEAK RATES (EPID L-2019-LLA-0115) DATED OCTOBER 20, 2020

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