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

June 26, 2020

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 281 AND 277 TO INCREASE ALLOWABLE MAIN STEAM ISOLATION LEAKAGE (EPID L-2019-LLA-0045)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 281 of Renewed Facility Operating License No. DPR-29 and Amendment No. 277 to Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The amendments are in response to your application dated March 5, 2019, as supplemented by letters dated May 23, 2019, July 22, 2019, February 24, 2020, and March 31, 2020.

The amendments revise the combined main steam isolation valve (MSIV) leakage rate limit for all four steam lines in Technical Specification (TS) 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' Surveillance Requirement (SR) 3.6.1.3.10; adds a new TS 3.6.2.6, ''Residual Heat Removal (RHR) Drywell Spray;'' and revise TS 3.6.4.1, ''Secondary Containment,'' SR 3.6.4.1.1.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Registe*r notice.

Sincerely,

/RA/

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-254 and 50-265

Enclosures:
1. Amendment No. 281 to DPR-29
2. Amendment No. 277 to DPR-30
3. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
License No. DPR-29

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated March 5, 2019, as supplemented by letters dated May 23, 2019, July 22, 2019, February 24, 2020, and March 31, 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 the license amendment and Paragraph 3.B of Renewed Facility Operating License No. DPR-29 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated March 31, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2020.06.26 08:59:16 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-29
and Technical Specifications

Date of Issuance: June 26, 2020

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
License No. DPR-30

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated March 5, 2019, as supplemented by letters dated May 23, 2019, July 22, 2019, February 24, 2020, and March 31, 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 the license amendment and Paragraph 3.B of Renewed Facility Operating License No. DPR-30 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance. Implementation of the amendment shall also include revision of the Updated Final Safety Analysis Report as described in the licensee's letter dated March 31, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2020.06.26 08:59:51 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-30 and Technical Specifications

Date of Issuance: June 26, 2020

ATTACHMENT TO LICENSE AMENDMENT NOS. 281 AND 277

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating areas of change.

REMOVE	INSERT
License DPR-29	License DPR-29
Page 4	Page 4
License DPR-30	License DPR-30
Page 4	Page 4

Technical Specifications

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	INSERT
3.6.1.3-8	3.6.1.3-8
------------	3.6.2.6-1
------------	3.6.2.6-2
3.6.4.1-2	3.6.4.1-2

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 281 are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

> The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined sets of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 249 as modified by License Amendment No. 259.

F. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979, with supplements dated November 5, 1980, and

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 277, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The license shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 244 and modified by License Amendment No. 254.

F. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979, with supplements dated

1 The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.10	Verify the leakage rate through each MSIV leakage path is ≤ 62.4 scfh for Unit 1 and 78 scfh for Unit 2 when tested at ≥ 25 psig, and the combined leakage rate for all MSIV leakage paths is ≤ 156 scfh for Unit 1 and 218 scfh for Unit 2 when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.2.6 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.2.6 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1 Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1 Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	-------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- C.1 Be in MODE 3.	 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.6.1	Verify each RHR drywell spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.6.2	Verify each drywell spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.6.3	Verify RHR drywell spray subsystem locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	----------------NOTE---------------- Not required to be met for 4 hours if analysis demonstrates one standby gas treatment (SGT) subsystem is capable of establishing the required secondary containment vacuum. ------------------------------------ Verify secondary containment vacuum is $\geq$ 0.10 inch of vacuum water gauge.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed, except when the access opening is being used for entry and exit.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	Verify the secondary containment can be maintained $\geq$ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate $\leq$ 4000 cfm.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.4	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 277 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By application dated March 5, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19064B369), as supplemented by letters dated May 23, 2019, July 22, 2019, February 24, 2020, and March 31, 2020 (ADAMS Accession Nos. ML19143A347, ML19203A176, ML20055E826, and ML20091H576, respectively), Exelon Generation Company, LLC (the licensee), requested changes to the technical specifications (TSs) for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed changes would revise the combined main steam isolation valve (MSIV) leakage rate limit for all four steam lines in TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' Surveillance Requirement (SR) 3.6.1.3.10; add a new TS 3.6.2.6, ''Residual Heat Removal (RHR) Drywell Spray;'' and revise TS 3.6.4.1, ''Secondary Containment,'' SR 3.6.4.1.1.

The supplemental letters dated May 23, 2019, July 22, 2019, February 24, 2020, and March 31, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 29, 2019 (84 FR 45543).

The July 22, 2019, supplemental letter submitted a request for a permanent exemption from the Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix J, Option B, Section III.A requirements in order to permit exclusion of MSIV leakage from the overall integrated leak rate Type A test measurement, and from Option B, Section III.B, requirements in order to permit

exclusion of the MSIV pathway leakage contributions from the combined leakage rate of all penetrations and valves subject to Type B and Type C tests. The NRC staff approved the exemption request on June 18, 2020 (ADAMS Accession No. ML20141L561).

2.0 REGULATORY EVALUATION

2.1 System Descriptions

2.1.1 MSIVs

The four main steam lines (MSLs), which penetrate the drywell, are automatically isolated by the MSIVs. There are two MSIVs on each steam line, one inside containment and one outside containment. The MSIVs are functionally part of the primary containment boundary and leakage through these valves provides a potential leakage path for fission products to bypass secondary containment and enter the environment as a ground level release.

2.1.2 RHR Drywell Spray

The RHR drywell spray system provides overpressure protection to the primary containment by quenching steam released to the drywell during a loss of coolant accident (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 a 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 Standby Gas Treatment (SGT) and Control Room (CR) Ventilation Systems

The secondary containment is a structure that encloses the primary containment including components that may contain primary system fluid. The safety function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a design basis accident (DBA) to ensure the control room operator and offsite doses are within the regulatory limits. There is no redundant train or system that can perform the secondary containment function should the secondary containment be inoperable.

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

During emergency conditions, the SGT system is designed to be capable of drawing down the secondary containment to a required vacuum within a prescribed time and continuing to maintain the negative pressure as assumed in the accident analysis. The leak tightness of the secondary containment together with the SGT system ensure that radioactive material is either contained in the secondary containment or filtered through the SGT system filter trains before being discharged to the outside environment via the elevated release point.

Both the SGT system and the CR ventilation system have a filter pack used to control radiation exposure during off-normal or accident conditions. The filter packs consist of the housing that contains the filters and absorber, the filters and absorber themselves, and any interconnecting ductwork between the filter elements. The SGT system filter pack treats the intentional release of primary and secondary containment atmosphere to the environs in the unlikely event of a DBA and thereby reduces exposure to the public and site personnel. The CR ventilation system filter pack provides protection from radiation exposure to allow control room access and occupancy for the duration of a LOCA with MSIV leakage at TS limits as the worst-case DBA.

2.2 Description of Proposed Changes

2.2.1 TS 3.6.1.3, ‘‘Primary Containment Isolation Valves (PCIVs)”

TS 3.6.1.3, ‘‘Primary Containment Isolation Valves (PCIVs),” SR 3.6.1.3.10 establishes and allowed leakage rates through each MSIV leakage path when tested at ≥ 25 pounds per square inch gauge (psig). The LAR proposes to increase this SR for each unit from ≤ 34 scfh to ≤ 62.4 scfh for QCNPS, Unit 1, and ≤ 78 scfh for QCNPS, Unit 2, and the combined leakage rate for all MSIV leakage paths when tested at ≥ 25 psig is increased for each unit from ≤ 86 scfh to ≤ 156 scfh for QCNPS, Unit 1, and ≤ 218 scfh for QCNPS, Unit 2.

2.2.2 TS 3.6.2.6, ‘‘Residual Heat Removal (RHR) Drywell Spray’’

New TS 3.6.2.6, “Residual Heat Removal (RHR) Drywell Spray” is proposed to be added to QCNPS, Units 1 and 2, TSs. LCO 3.6.2.6 would require two RHR drywell spray subsystems to be operable and has an Applicability of Modes 1, 2, and 3. LCO 3.6.2.6 would have three Actions which are:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1 Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1 Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in Mode 3. <u>AND</u> C.2 Be in Mode 4.	12 hours 36 hours

Three SRs, as shown in the following table, would be associated with new LCO 3.6.2.6.

SURVEILLANCE		FREQUENCY
SR 3.6.2.6.1	Verify each RHR drywell spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.6.2	Verify each drywell spray nozzle is unobstructed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.2.6.3	Verify RHR drywell spray subsystem locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

2.2.3 TS 3.6.4.1, ''Secondary Containment''

The LAR proposes to add a Note to SR 3.6.4.1.1 that would allow the secondary containment vacuum limit to not be met for a short duration during the period provided an analysis demonstrates that one SGT subsystem remains capable of establishing the required secondary containment vacuum. The proposed Note states that "Not required to be met for 4 hours if analysis demonstrates one standby gas treatment (SGT) subsystem is capable of establishing the required secondary containment vacuum" for the SR to "Verify secondary containment vacuum is ≥ 0.10 inch of vacuum water gauge.

2.3 Regulatory Requirements and Guidance

Under 10 CFR 50.90, whenever a holder of a license wishes to amend the license, including technical specifications in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR § 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be "reasonable assurance" that the activities at issue will not endanger the health and safety of the public, and will comply with the NRC's regulations.

2.3.1 Alternate Source Term (AST)

Licensees using the AST are evaluated against the dose criteria specified in 10 CFR 50.67, "Accident source term," (b)(2):

- 10 CFR 50.67(b)(2)(i) requires that an individual located at any point on the boundary of the exclusion area (EAB) for any 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 roentgen equivalent man (rem) (0.25 sieverts (Sv)) total effective dose equivalent (TEDE).

- 10 CFR 50.67(b)(2)(ii) requires that an individual located at any point on the outer boundary of the low population zone (LPZ), who is exposed to the radioactive cloud resulting from the postulated fission product release would not receive a radiation dose in excess of 25 rem (0.25 SV) TEDE during the entire period of its passage.

- 10 CFR 50.67(b)(2)(iii) requires that adequate radiation protection be provided to permit access to and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 5 rem (0.05 Sv) TEDE for the duration of the accident.

As stated in Section 3.1 of the updated final safety analysis report (UFSAR), based on the understanding of the intent of the proposed criteria current at the time of operating license application, QCNPS, Units 1 and 2, conforms with the intent of the Atomic Energy Commission General Design Criteria (GDC) for Nuclear Power Plant Construction Permits. As the GDC were finalized, the requirements were placed in Appendix A to 10 CFR Part 50. Under 10 CFR Part 50, Appendix A, GDC, Criterion 19 – “Control room,” 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 LOCAs. 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) TEDE as defined in 10 CFR 50.2, for the duration of the accident.

As stated in NRC Regulatory Issue Summary (RIS) 2006-04, “Experience with Implementation of Alternative Source Terms,” dated March 7, 2006 (ADAMS Accession No. ML053460347), any licensee who chooses to reference AEB 98-03 assumptions should provide an appropriate justification that the assumptions are applicable to their particular design.

For licensees using the AST in their radiological consequence analyses, the NRC staff uses the NRC regulatory guidance:

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, dated July 2000 (ADAMS Accession No. ML003734190).

- NUREG-0800, SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, dated March 2007 (ADAMS Accession No. ML070190178).

- Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, dated July 2000 (ADAMS Accession No. ML003716792).

2.3.2 Environmental Qualification

Under 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” licensees must establish a program for qualifying electric equipment that is important to safety as defined in 10 CFR 50.49(b). Subsection 50.49(e)(1) requires that the time-dependent temperature and pressure at the location of the electric equipment important to safety be established for the most severe DBA during and following which this equipment is required to remain functional. Subsection 50.49(e)(2) requires that humidity during DBAs be considered. Subsection 50.49(e)(4) requires that the radiation environment 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 design basis accident 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. Subsection 50.49(b)(2) requires qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs 50.49(b)(1)(i)(A)-(C) by the safety-related equipment.

2.3.3 TSs

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

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. The remedial actions must provide the requisite "reasonable assurance" of safety and compliance.

The regulation at 10 CFR 50.36(c)(3), requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The NRC staff's guidance for review of TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," dated March 2010 (ADAMS Accession Number ML100351425).

2.3.4 Human Factors

NUREG-0800, Chapter 18, "Human Factors Engineering," Revision 3 (ADAMS Accession No. ML16125A114), contains guidance for the NRC staff on human factors.

NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, and NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1 (ADAMS Accession No. ML12324A013 and ML072640413, respectively), contain guidance on state-of-the-art human factors principles.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences

To demonstrate that the performance of various plant safety systems designed to mitigate the postulated radiological consequence at QCNPS, Units 1 and 2, will remain adequate after implementing the MSIV leakage rate limit increases in the requested TS changes, the LAR included a revision to the postulated radiological consequence analysis of the design basis LOCA.

Attachment 1 and Enclosure B of the LAR (ADAMS Accession No ML19064B371), as supplemented by the response to RAIs dated March 31, 2020 (ADAMS Accession No. ML20091H576), provide the results of the revised design basis LOCA radiological analysis to demonstrate compliance with 10 CFR 50.67 for the CR, EAB, and LPZ doses, and 10 CFR Part 50, Appendix A, GDC 19, for CR dose.

The revised design basis LOCA radiological analysis was performed using an NRC radiological consequence computer code, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, described in NUREG/CR-6604, "A Simplified Model for RADionuclide Transport and Removal And Dose Estimation" (ADAMS Accession No. ML15092A284). The RADTRAD code, developed by the Sandia National Laboratories for NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performed independent confirmatory dose evaluations, as needed, using the RADTRAD code. The results of the evaluations performed by the licensee (QDC-0000-N-1481 Revision 4 provided in Attachment 2 to the March 31, 2020 letter), as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 below.

Table 1
QCNPS Units 1 and 2 Bounding LOCA Radiological Consequences Expressed as TEDE (1) (rem)

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (rem) Receptor Location		
	CR	EAB (2)	LPZ (3)
Containment Leakage	0.236	0.331	0.686
ESF Leakage	0.00895	0.00537	0.099
MSIV Leakage	2.92	16.6	2.94
Reactor Building Shine	0.143	0	0
External Cloud Shine	0.359	0	0
CR Filter Shine	Negligible	0	0
Total Dose	3.66	16.9	3.72
Acceptance Criteria	5	25	25

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

The revised design basis LOCA radiological analysis is based, in part, on the current design basis LOCA radiological analysis approved by the NRC in License Amendment Nos. 233 and 229, "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Source Term Methodology

(TAC Nos. MB6530, MB6531, MB6532, MB6533, MC8275, MC8276, MC8277 and MC8278)," dated September 11, 2006 (ADAMS Accession No. ML062070290). The amendment adopted full implementation of the AST methodology. In the NRC staff's safety evaluation (SE) issued with the amendment approving full implementation of the AST methodology, the NRC staff indicated that it had concerns regarding the use of AEB 98-03 used in the QCNPS, Units 1 and 2, current licensing basis (CLB). At that time, the NRC staff based its approval of the AST license amendment request (LAR), in part, upon additional conservatism in the MSIV leakage model, which is no longer present in the proposed LAR:

> The NRC staff expressed a concern that the removal through aerosol settling was overestimated by modeling two settling volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping and at a later time considering that the larger and heavier aerosols would have already settled out of the main steam line atmosphere in upstream sections of piping. However, as stated above, Exelon [the licensee] did not credit any reduction in drywell pressure or the MSIV leakage rate after 24 hours. Leakage rates were assumed to be held constant for the entire duration of the accident for conservatism. Given this information, the NRC staff finds the Dresden and Quad Cities main steam line aerosol settling model to be reasonably conservative.
>
> The NRC staff acknowledges that aerosol settling is expected to occur in the main steam line piping but because of recent concerns with aerosol sampling and its characteristics used in AEB-98-03 and lack of further information, the NRC staff is concerned with how much deposition (i.e., what settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, but has applied additional conservatism (i.e. 40th percentile settling velocity, constant MSIV leakage for the entire duration of the accident) to address the NRC staff's concern about the applicability of the AEB-98-03 methodology to Dresden and Quad Cities. The NRC staff further acknowledges that the estimate of the fraction of the aerosol that leaks to the environment is uncertain because of phenomenological uncertainties concerning the environment the aerosol encounters in the various volumes assumed by Exelon.

The changes requested in this LAR modify several assumptions and inputs used to model the MSIV leakage pathway after a design basis LOCA and, thus, removed additional conservatism used by the NRC staff to find the CLB analysis reasonably conservative. For example, in the revised analysis, the RHR drywell spray system is credited for the reduction of airborne activity in the drywell by scrubbing radionuclides from the drywell air space, mitigating the consequence of the postulated design basis LOCA. The particulate iodine deposition from the RPV nozzle to the inboard MSIV, elemental iodine deposition from the RPV nozzle to the inboard MSIV, elemental iodine between inboard and outboard MSIVs, and MSIV leakage after 24 hours to 50% of the maximum leakage are credited. To model the effect of mixing, the MSIV flow rate used in the RADTRAD model was decreased by calculating a new leak rate based on the combined volumes of the drywell and suppression chamber.

3.1.1 Accident Source Term/Core Inventory

The licensee evaluated the CR, EAB, and LPZ radiological consequences due to the design basis LOCA. In accordance with RG 1.183 guidance, the inventory of fission products in the reactor core was determined based on the maximum full power operation of the core (3,016

megawatts thermal (MWt)). This maximum full power operation is 102% of the 2,957 MWt-rated power, which includes margin for measurement uncertainty using the Framatome ATRIUM 10XM core inventory having a core average exposure of 39 gigawatt-days per metric ton of uranium (GWD/MTU). The chemical form of radioiodine released into the containment is assumed to be 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide. Except for elemental and organic iodine and noble gases, the remaining fission products are assumed to be in particulate form. The isotopic core inventory in Curies (Ci) for the Framatome fuel design is provided in Table 1 and Section 5.3, "Containment Leakage Model Parameters" of the LAR. The LAR stated in Section 8.2, "Conclusions," that the Westinghouse SVEA-96 Optima 2 fuel type was being phased out, it was the most limiting fuel type with respect to dose consequences, and the design basis LOCA doses will continue to be based on the Westinghouse SVEA-96 Optima 2 fuel type until both QCNPS, Units 1 and 2, no longer contained that fuel type.

The NRC staff found the AST/core inventories; BWR core inventory fractions and release timings; the QCNPS, Units 1 and 2, maximum rated thermal power with uncertainty; and the chemical forms and percentages of radioiodines released into the containment were based on bounding fuel inventories and enrichments, and are consistent with the NRC guidance in RG 1.183 and, therefore, are acceptable.

3.1.2 MSIV Leakage

The LAR stated that the MSIVs are postulated to leak at a total design leakage rate of 250 scfh (standard cubic feet per hour) for QCNPS, Unit 1, and 350 scfh for QCNPS, Unit 2. QCNPS, Unit 2, is analyzed for a higher leakage rate than QCNPS, Unit 1 because the QCNPS, Unit 2, MSIV to CR ground-level release χ/Q values are lower than the χ/Q values for QCNPS, Unit 1. The radiological consequences from a postulated MSIV leakage from crediting RHR drywell sprays to reduce isotopes escaping containment is analyzed and combined with the radiological consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the design basis LOCA. This is summarized in Attachment 1, Table 3-2: LOCA Dose Consequence Summary of the LAR, as revised in response to a request for additional information in Attachment 1, Table RAI-2a Revised Bounding LOCA Dose Consequence Summary of the RAI response dated March 31, 2020.

The LAR stated that all MSL piping sections between the RPV nozzle and outboard MSIVs used in the MSIV leakage release paths are assumed to remain intact and can perform their safety function during and following a safe shutdown earthquake (SSE). Based on the structural integrity and functional performance of the MSL piping up to the outboard MSIV to withstand the SSE, the horizontal pipe surface area and volume is credited in the aerosol removal calculation. A total MSIV leakage of 250 scfh for QCNPS, Unit 1, and 350 scfh for QCNPS, Unit 2, was assumed to occur:

1) 100 scfh for QCNPS, Unit 1, and 125 scfh for QCNPS, Unit 2, through the steam line with the "failed" MSIV. The deposition removal of aerosol in the horizontal pipe, and the deposition removal of elemental iodine in both the horizontal and vertical pipes, are credited in the steam line between the RPV nozzle and outboard MSIV.

2) 100 scfh through first intact steam line for QCNPS, Unit 1, and 125 scfh for QCNPS, Unit 2. The deposition removal of aerosol in the horizontal pipe, and the deposition removal of elemental iodine in both the horizontal and vertical pipes, are credited in the steam line between the RPV nozzle and outboard MSIV.

3) 50 scfh through the second intact steam line for QCNPS, Unit 1, and 100 scfh for QCNPS, Unit 2. The deposition removal of aerosol in the horizontal pipe, and the deposition removal of elemental iodine in both the horizontal and vertical pipes are credited in the steam line between the RPV nozzle and outboard MSIV.

4) 0 scfh through the fourth steam line for both units.

3.1.3 LOCA Analysis

A summary of the changes and the reason for each change to the methodology and inputs of the revised design basis LOCA radiological analysis compared to the CLB analysis are provided in Attachment 1, Table 3-1: Summary of LOCA Analysis Revisions of the LAR, as revised in Attachment 1, Table RAI-2a Revised Bounding LOCA Dose Consequence Summary of the RAI response.

Attachment 1, Table 3-2: LOCA Dose Consequence Summary in the LAR showed the CR dose decreased as compared to the CLB CR dose primarily from crediting RHR drywell sprays to reduce isotopes escaping containment.

The revised design basis LOCA radiological analysis credits the reduction in the containment leakage and MSIV leakage after 24 hours to 50% of the maximum leakage with respect to RADTRAD flowrates.

The radiological consequence analysis for the postulated design basis LOCA in the CLB is based on the NRC-approved use of an AST described in QCNPS, Units 1 and 2, UFSARs, Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment.” The results of CR, EAB, and LPZ doses were re-analyzed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at QCNPS, Units 1 and 2, will remain adequate after the implementation of the increased MSIV leakage rate limits and to credit the RHR drywell spray system for accident mitigation to support the TS changes requested in the LAR.

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

- Containment leakage
- ESF leakage
- MSIV leakage

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

- Reactor building shine
- External cloud shine
- CR filter shine

Revision 4 of QDC-0000-N-1481, Table RAI-2a, showed slight dose increases in the containment leakage (for the CR, EAB and LPZ) and the reactor building shine (for the CR) pathways compared to the CLB. No changes in doses were observed in the engineered safety function (ESF) and MSIV leakage, external cloud shine, and CR filter shine pathways from the CLB. The NRC staff concluded based on the licensee's analysis that the other release pathways and LOCA dose contributors were small compared to the MSIV leakage dose. Therefore, the NRC staff focused its review of the MSIV leakage increase proposed in the LAR by reviewing the MSIV leakage release pathway which is the most significant contributor to the design basis LOCA dose.

The licensee calculated and submitted the results of the CR, EAB, and LPZ doses and provided the major assumptions and parameters used in its dose calculations in the LAR. The LAR stated that after implementation of the increased MSIV leakage rate limits, crediting the RHR drywell spray system for accident mitigation, the TS changes, the use of an AST, and the existing ESF systems at QCNPS, Units 1 and 2, the radiological consequences of the postulated design basis LOCA at the EAB, LPZ, and in the CR will meet the dose criteria specified in 10 CFR 50.67 and in 10 CFR Part 50, Appendix A, GDC 19.

The revised design basis LOCA radiological analysis provided in the LAR removed credit in the CLB for the Powers' deposition model from NUREG/CR-6189 "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (ADAMS Accession No. ML100130305) in the drywell for the particulate (aerosol) deposition/plateout in containment. This consisted of approximately 32,430 square feet (ft^2) of surface area available in the drywell for deposition and plateout of aerosols not credited in the RADTRAD model. The only surface area credited for deposition and plateout was upstream of the outboard MSIVs in three of the four MSLs which has a surface area of 237 ft^2. Section 5.8, "Changes Between Revision 2 and Revision 3" of the LAR stated that credit was removed for the Powers' deposition model due to an error notice posted to the RADTRAD Industry Users Group's website (radtrad.com) indicating that it that may underestimate doses for BWRs. The NRC staff found this acceptable because the Powers' deposition model was not used in the revised design basis LOCA radiological analysis.

Drywell Spray Assumptions in the LOCA Model

RG 1.183, Appendix A RP 3.3, states that reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP may be credited. The LAR analysis based the credited spray removal on a spray pump volumetric flow rate of 2,352 gallons per minute (gpm). The analysis assumed that the spray would be initiated by manual action 10 minutes post-accident with an assumed termination at 4 hours and a fall height of 11.41 meters (m) (37.43 feet (ft)) based on the difference in elevations between the lower drywell spray header and the bottom of the drywell floor.

The NRC staff reviewed the QCNPS, Units 1 and 2, UFSAR, Section 6.2.2, "Containment Heat Removal Systems," to determine if the containment spray systems were designed to provide a reduction in airborne activity consistent with SRP Section 6.5.2. Based on the NRC staff's review, it appeared that the spray systems were designed for pressure reduction and not specifically for reducing airborne radioactivity. The NRC staff noted that containment spray design requirements regarding the ability to reduce airborne radioactivity were discussed in LAR Enclosure B, Section 2.1.3, "Reduction in Airborne Activity Inside Containment," in a comparison between SRP Section 6.5.2 review items and how those items are addressed in the

revised analysis. To address this concern, the NRC staff issued RAI No. ACRB-RAI-1A requesting additional information describing how the design characteristics of the containment spray systems provide a reduction in airborne activity in accordance with SRP 6.5.2, as discussed in Enclosure B, Section 2.1.3 of the LAR, will be incorporated into the QCNPS, Units 1 and 2, UFSAR.

The letter dated March 31, 2020 responded to RAI No. ARCB-RAI-1A and explained the QCNPS, Units 1 and 2, UFSAR will be updated in accordance with 10 CFR 50.71(e) as part of implementation of the approved amendment. The response provided the following summary of the proposed changes:

- Sections 6.0.1.2, 6.2.2.1, and 6.3.1.2 will be updated to include how drywell spray aids in removal of airborne fission products.

- Section 6.2.1.3.2.2 will be revised to remove statements about drywell spray not being necessary. The statement about there being no time requirement for initiation of the containment cooling system will also be removed.

- Section 6.5.2 will be revised to summarize the design characteristics of the drywell spray system that impact its ability to provide a reduction in airborne activity. The level of detail will be similar to that included in the table in Section 2.1.3 of QDC-0000-N-1481. These design characteristics and includes meeting the requirements of ANS/ANSI 56.5 as it relates to calculation of airborne fission product removal following a LOCA such as geometry, physical features, flow characteristics, and mixing considerations as described in Standard Review Plan Section 6.5.2.

The NRC staff reviewed the calculation of the particulate removal coefficient as documented in Enclosure B, Section 7.11, "Spray Calculations," page 64 of the LAR. Based on this review, it appeared that the spray drop fall height of 11.41 m (37.43 ft) was determined by the difference in elevations between the lower drywell spray header and the bottom of the drywell floor. This method did not appear to consider the obstructions that are present in the drywell, which could reduce the effective spray drop fall height. In addition, the analysis assumed a spray flow rate of 2,352 gpm. As with spray drop fall height, obstructions in the drywell could reduce the effective spray flow rate available for reducing airborne radioactivity. The NRC staff noted that both the unobstructed free fall height and spray flow rate are important factors in determining the ability of the containment sprays to effectively reduce airborne radioactivity. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," Section H (ADAMS Accession No. ML063480542), discusses the issue of obstructions interfering with the effectiveness of sprays as follows:

> H. Droplet-Structure Interactions
>
> Reactor containment buildings are not simple, open volumes. Immediately below spray headers there is often a substantial open space. But, eventually, falling drops begin to encounter equipment, structures and operating floor of the reactor. The drywells of Mark I containments are well-known for the congestion that can interfere in the free fall of water droplets.

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

To address the concern on spray fall height, the NRC staff issued RAI No. ARCB-RAI-1B requesting a justification for the use of the assumed spray fall height of 11.41 m (37.43 ft) in the determination of the particulate removal coefficient which apparently did not consider obstructions present in the drywell.

The letter dated March 31, 2020, responded to RAI No. ACRB-RAI-1B and stated the spray removal coefficient used in Revision 3 of QDC-0000-N-1481 for a decontamination factor (DF) ≤ 50 is 15.0 hour^{-1} (hr^{-1}). The response stated that this spray removal coefficient value was based in part on a spray fall height calculated as the difference between the lower spray header elevation (607 ft (')-3 inches (")) and the bottom of drywell elevation (569'-10"). The calculated spray removal coefficient value of 20.44 hr^{-1} was reduced to 15.0 hr^{-1} when input into the RADTRAD model as a conservatism. Consistent with SRP Section 6.5.2, this reduced value of 15.0 hr^{-1} for DF ≤ 50 was further reduced by a factor of 10 to 1.5 hr^{-1} when a DF of 50 was reached in the RADTRAD model.

Attachment 2, Revision 4 of QDC-0000-N-1481, Section 7.11, provided in the letter dated March 31, 2020, asserted the calculated spray removal coefficient was conservative as compared to the methodology used for the implementation of the Nine Mile Point Unit 1 (NMP1) (ADAMS Accession No. ML073230597) and Oyster Creek (OC) (ADAMS Accession No. ML050940234) ASTs. In the NMP1 and OC ASTs, specific reductions were taken in the calculation based on obstructions in the drywell or blocked nozzles that may impede flow. In addition to the equipment installed in the drywell, the obstructions included two floors of grating between the spray headers and the bottom of the drywell: one between the two spray headers at elevation (614'-7¼") and one below the lower spray header at elevation (592'-11¾"). The Revision 4 fall height calculation was based on the drywell floor elevation (579'-10") rather than the Revision 3 bottom of drywell elevation (569'-10") and the overall spray removal coefficient was calculated using the Revision 4 lower spray header elevation (592'-11¾").

The response explained that since both elevations of spray nozzles (upper at 628'-8" and lower at 607'-3") will be available following a LOCA, the average fall height between these two elevations could have been used to calculate the fall height but using the lower header elevation provides some additional conservatism. The licensee concluded that in conjunction with the conservatisms in the spray flow rate discussed in its response to RAI No. ARCB RAI-1C (evaluated in this section of the SE below), the overall spray removal coefficient was conservative as compared to the methodology used for NMP1 and OC, which reduced the average spray header fall height to account for obstructions including grating and equipment.

The response to RAI No. ACRB-RAI-1B, referred to the methodology used for NMP1, OC, and Nine Mile Point Unit 2 (NMP2) AST implementation which made specific reductions in the spray removal coefficient calculation based on obstructions in the drywell or blocked nozzles that may impede flow. For example, the NRC staff's SE associated with the implementation of the NMP1 AST states, in part, the following:

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

The response referred to the OC AST which applied the three-dimensional modeling of the drywell. This modeling resulted in a 33.3 % reduction in fall height to account for obstructions and a 33.3% reduction in flow rate based on a modular accident analysis program analysis. The NRC staff's SE associated with the OC AST states, in part, that "Drywell congestion explicitly addressed by reduced spray flow and fall height."

To address the concern related to spray flow rate, the NRC staff issued RAI No. ARCB-RAI-1C requesting a justification for the use of the assumed spray flow rate of 2,352 gpm in the determination of the particulate removal coefficient which apparently did not consider obstructions present in the drywell.

Attachment 2, Revision 4 of QDC-0000-N-1481, Section 7.11, explained the assumed drywell spray flow rate of 2,352 gpm was based on 160 drywell spray nozzles providing 14.7 gpm each. Each ring header contained 160 nozzles spaced around the drywell. The response further explained that the spray flow rate assumed in the analysis was based on only a single header providing flow even though both headers can be supplied simultaneously by a single RHR pump. In addition, UFSAR Section 6.2.1.3.3 states, in part, that the design basis drywell spray flow rate is 4,750 gpm and the wetwell spray flow rate is 250 gpm. TS SR 3.6.2.3.2 specifies that each required RHR pump develops a flow rate greater than or equal to 5,000 gpm while operating in the suppression pool cooling mode. This TS SR for RHR pump flow rate is substantially greater than the drywell spray flow rate of 2,352 gpm (47% of the TS SR 3.6.2.3.2 RHR pump flow rate) assumed in the calculation of the spray removal coefficient.

The response referred to the NMP2 AST methodology used to determine the flow rate reduction due to potential nozzle blockage. This methodology assumed that certain percentages of nozzles were blocked using survey data. Based on the NMP2 AST methodology, the licensee determined that the difference in the overall reduction was negligible and will still lead to a spray removal coefficient greater than 15.0 hr^{-1}.

The NRC staff notes that the overall spray removal coefficient considers both spray fall height and spray flow rate and there was no specific reduction taken in the spray fall height to consider obstructions in the drywell. However, based on a review of the responses to RAI Nos. ACRB-RAIs-1A, -1B, and -1C, the NRC staff finds the overall spray removal coefficient of 15.0 hr^{-1} for a DF ≤ 50 and 1.5 hr^{-1} for a DF > 50 is acceptable, because it addressed the overall impact of spray fall height and spray flow rate from obstructions that are present in the drywell which could reduce the effective spray drop fall height and effective spray flow rate available for reducing airborne radioactivity.

Drywell Spray Credit in the LOCA Model

<u>Crediting Iodine Removal in MSL Piping</u>

RG 1.183, Appendix A, Section 6.3 states, in part, that the "Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the

outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis."

Attachment 1, Table 3-1, "Summary of LOCA Analysis Revisions," of the LAR presented changes to the CLB for the revised design basis LOCA radiological analysis. One of the proposed changes involves a change to the elemental iodine removal credited in the main steam lines (MSLs). The CLB credits elemental iodine removal in the two intact steam lines but not in the line with the failed MSIV. The LAR proposed to substantially increase the elemental iodine removal in the MSLs between the reactor pressure vessel (RPV) and the outboard MSIV by crediting elemental removal in the line with the assumed failed MSIV and by increasing the removal in the previously credited volumes from 50% to up to about 98%.

From the NRC staff's examination of Enclosure B and Section 7.3, "Main Steam Line Volumes & Surface Area for Plateout of Activity," page 54 of the LAR, some discrepancies in the tabulated data and parameter values applied as parameters in the revised LOCA radiological analysis were observed:

- Table 1B, "Rate Constant (λ_s) for Aerosol Settling In Main Steam Piping," page 77. The 40^{th} percentile settling velocity given as "0.0081 m/s" should be "0.00081 m/s."

- Table 20, "MSIV Failed & Intact Steam Line Volumes for Elemental Iodine Removal Efficiency Calculation," page 95. The calculated volume for "D" (Volume V_4) given as "4.33 m^3" should be "4.64 m^3." The calculated volume of "E" (Volume V_5) of "4.33 m^3" should be "1.39 m^3."

- Table 26, "Elemental Iodine Deposition Rate - Intact Steam Line Volume $V_{4,}$" page 98. The Main Steam Line Total Surface Area given as "10.07 m^2" should be "12.35 m^2." As a result, the Elemental Iodine Removal Rates (hr^{-1}) and Elemental Iodine Deposition Efficiencies for all listed post-LOCA times in Table 26 are impacted.

- Table 31, "Net Elemental Iodine Removal Efficiency - Intact Steam Line Volume $V_{4,}$" page 101. As a result of Table 26 observed discrepancies, the Elemental Iodine Deposition Efficiencies, Elemental Iodine Resuspension Efficiencies, and Elemental Net Deposition Efficiencies (%) for all listed post-LOCA times in Table 31 are impacted.

- As a result of the Table 31 observed discrepancies, the RADTRAD model input parameter values for elemental iodine are impacted.

To address the concern on discrepancies in the tabulated data and parameter values applied as inputs in the revised design basis LOCA radiological analysis, the NRC staff issued RAI No. ARCB-RAI-2 requesting that the observed discrepancies described above be addressed and an evaluation of the impact on the calculated CR, EAB, and LPZ doses in the revised design basis LOCA radiological analysis be conducted.

The letter dated March 31, 2020, provided a response to RAI No. ARCB-RAI-2 and made the following changes, some of which resulted in RADTRAD recalculations. Attachment 1, Table 1B, corrected the typographical error of the 40^{th} percentile settling velocity which was not used in the RADTRAD models and had no impact on the calculation result. Typographic errors were corrected in the note to Table 20 to the calculated volumes for "D" (Volume V_4) and "E"

(Volume V_5). These volumes were typographic errors in the note; the correct volumes were provided in Table 20 and were used to calculate the values used in the RADTRAD models and had no impact on the calculation result. The surface area ("B") was corrected and used in Table 26 for all timesteps and the corresponding Elemental Iodine Removal Rate ("D") and Elemental Iodine Deposition Efficiency ("E") were updated as well. This change increased the removal of elemental iodine which resulted in a slight decrease in the calculated doses. The Elemental Iodine Deposition Efficiency ("A") was updated in Table 31 for all timesteps to the corrected values from Table 26. The resulting Elemental Iodine Net Deposition Efficiency ("C") was updated for each timestep. The response stated that the net deposition efficiency from Table 31 was used as input into the MSIV leakage RADTRAD cases, and 10 updated MSIV leakage cases were run, 5 for each fuel type, using the revised values.

The response to RAI No. ACRB-RAI-2 stated the above discrepancies were captured in the vendor's corrective action program that prepared the calculation. The response also stated, as part of the associated corrective action, a separate review was performed to ensure there were no other discrepancies that would affect the calculated dose results. Based on this review, a containment leakage pathway from the unsprayed drywell volume to the RB was added for the containment leakage RADTRAD cases provided in Attachment 2, revised Figure1 of QDC-0000-N-1481 Revision 4. The updated bounding design basis LOCA doses, including the additional containment leakage pathway and corrected discrepancies in Attachment 1, Table RAI-2a, "Revised Bounding LOCA Dose Consequence Summary," displayed the resulting slight increase in calculated doses in the Containment Leakage and Reactor Building Shine Post-LOCA Activity Release Paths and the CR due to the additional containment leakage pathway.

Based on a review of the response to RAI No. ACRB-RAI-2, the NRC staff finds the changes described above acceptable because the typographical errors on the 40th percentile settling velocity and calculated MSIV Failed & Intact Steam Line Volumes (V4 and V5) for Elemental Iodine Removal Efficiency Calculation in Attachment 1, Table 1B and the note in Table 20; the MSL Total Surface Area, Elemental Removal Rates, and Elemental Iodine Deposition Efficiencies in Table 26; and the Elemental Iodine Deposition Efficiencies, Elemental Iodine Resuspension Efficiencies, and Elemental Net Deposition Efficiencies for all listed post-LOCA times in Table 31 and those RADTRAD model input parameters for elemental iodine were corrected in the updated bounding design basis LOCA doses.

The NRC staff's evaluation on the MSIV leakage model that assumes the same 40th percentile settling velocity, as in the CLB, but with credit for sprays, including the recalculated bounding design basis LOCA doses provided in response to RAI No. ACRB-RAI-2 is discussed in Sections 3.1.4 and 3.1.5 of this SE.

Aerosol Removal in Steam Lines with Sprays Credited

RG 1.183, Appendix A, Section 6.3, states, in part, that the "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." Section 6.5 states, in part, that the "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 a safe shutdown earthquake and that the amount allowed will be evaluated on an individual case basis."

SRP Section 15.0.1 states, in part, that "Independent calculations should be performed as necessary to conclude, with reasonable assurance, that the applicant's analyses are acceptable."

Attachment 1, "Evaluation of Proposed Changes," page 16 of the LAR states, in part:

> The approved main steam line aerosol removal model does not include deposition by thermophoresis, diffusiophoresis, or flow irregularities.
>
> Therefore, it is reasonable to consider the use of aerosol removal by sprays and aerosol removal in the main steam lines as independent removal mechanisms because they rely on different physical mechanisms except for diffusiophoresis. However, neither the containment spray model nor the aerosol removal in main steam lines model consider removal by diffusiophoresis which confirms the modeling is conservative with respect to the experimental data.

Enclosure B, Section 5.8, "Changes Between Revision 2 and Revision 3," page 43 of the LAR, states, in part, that the "Drywell spray meets the requirements in NUREG-0800 Section 6.5.2 as demonstrated in Section 2.1.3 and has been previously accepted for Nine Mile Point Units 1 and 2, Oyster Creek, and Hatch."

The NRC staff notes that the AST applications cited above with credited drywell sprays were previously accepted on an individual case basis that included considerations of the particular design and under different conditions, such as credit applied for the condenser, lower MSIV leakage rates and decontamination factors, and a "penalty" applied for sedimentation (aerosol settling) to account for the recognition that the sprays preferentially remove large particles in primary containment. For example, in the NMP2 AST application, an aerosol settling velocity of 0.000066 m/s (compared to an aerosol settling velocity of 0.00081 m/s proposed in the QCNPS LAR) was applied to reflect the spray removal credited in the NMP2 containment, and to address the NRC staff's concerns regarding the use of AEB 98-03. In its approval of the NMP2 application, the NRC staff found this value to be sufficiently conservative (along with other conservatisms) to reflect the effectiveness of the sprays.

NUREG/CR-5966 provides details on how sprays impact aerosols. This guidance document indicates that the sprays shift the sizes of aerosols in the containment towards those that are removed most slowly (the mean aerosol size decreases as the sprays operate). Estimates of aerosol deposition in the steam lines is determined using, in part, Equation 5 of AEB 98-03. Equation 5 provides the aerosol settling (and thus the aerosol deposition) in the steam line and indicates that the aerosol settling is proportional to the square of the diameter of the aerosols. Because the sprays shift the size of the aerosols to smaller sizes, the aerosols settling in the steam lines would decrease due to these smaller diameter aerosols.

The LAR proposes to credit sprays to remove fission products following a design basis LOCA, but it does not appear to adjust the MSL aerosol deposition from the impact of the sprays in the revised LOCA radiological analysis. Enclosure B, Table 1B, "Rate Constant (λ_s) for Aerosol Settling in Main Steam Piping," page 77 of the LAR shows the same 40th percentile aerosol settling velocity (0.00081 m/s) in all control volumes as used in the CLB with no credit for sprays. This is non-conservative when applying credit for sprays and considering the additional

conservatism in the CLB when sprays were not credited, which would be removed through this LAR. The sprays change the aerosols on a time-dependent basis through each control volume that impacts its removal in the MSLs.

From the NRC staff's examination of the submitted information, it appears that the revised design basis LOCA radiological analysis considers the aerosol removal by sprays and aerosol removal in the MSLs as independent removal mechanisms. The NRC staff notes that regardless of the specific removal mechanisms involved, larger aerosol particles in the containment atmosphere will be the preferentially removed, thereby making subsequent removal by deposition in downstream piping more challenging.

To address the concern on aerosol removal in the MSLs with sprays credited, the NRC staff issued RAI No. ARCB-RAI-3 requesting justification as to why the proposed aerosol settling velocity and model to credit sprays in the QCNPS, Units 1 and 2, design is consistent with RG 1.183, Revision 0, and to include sufficient technical detail to enable the NRC staff to perform an independent assessment on this aerosol settling velocity and model, and the subsequently calculated CR, EAB, and LPZ doses.

The letter dated March 31, 2020, responded to RAI No. ARCB-RAI-3 and asserted inherent conservatisms in the design basis LOCA radiological consequence assessment and presented sensitivity analyses to show the uncertainty introduced by the drywell spray effects on the aerosol deposition model.

The response to RAI No. ARCB-RAI-3, stated that the QCNPS, Units 1 and 2, CLB includes a number of conservatisms included in the LOCA dose consequence assessment that were credited as part of the approval of the AST amendment, whose design basis was provided by QDC-0000-N-1481 Revision 1. The response stated both the CLB and the revised LOCA AST dose analysis assume the drywell is the source of MSIV leakage in accordance with the NRC guidance in RG 1.183, and that it is appropriate to consider radionuclide removal mechanisms in the drywell before release via the MSIV leakage pathway. The conservatisms asserted in the response are evaluated in Section 3.1.5 of this SE.

3.1.4 Licensee's Sensitive Study - Use of 20-Group Method

The response to RAI No. ACRB-RAI-3 stated that a simplified model was developed using first principles as identified in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays." 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 QDC-0000-N-1481, 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 sprays at 4 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 response stated that the analysis 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 (GSD) of 2. The aerosol mass was calculated for each group independently with no

consideration of particles interacting with one another (i.e. not accounting for agglomeration). Table RAI-3a: Drywell Particle Size summarizes the results of the 20-group particle size distribution in the drywell. Figure RAI-3a: Time-Dependent Aerosol Particle Size Distribution shows the time-dependent nature of the aerosol particle size distribution and shows the effect of the drywell spray in reducing the size of the particles in the model. The particle size and settling velocity distributions were then used to recalculate the aerosol removal factors using the equation in Section 7.4.1 of QDC-0000-N-1481 summarized in Table RAI-3d: 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 in Table RAI-3e: Sensitivity Study Results. The sensitivity analyses use the QCNPS, Unit 1, RADTRAD model inputs. The response stated that the relative change in the calculated doses are expected to be similar for the QCNPS, Unit 2, RADTRAD model inputs.

The response stated that a total of 7 sensitivity cases were performed from various combinations of breathing rate, MSIV impaction, and condenser holdup/aerosol deposition by varying the Base Sensitivity Case ("base case"). The base case is the QCNPS, Unit 1, QDC-0000-N-1481, Revision 4, model including the nodalization adjustments and the revised aerosol removal factors. The QCNPS, Unit 1, QDC-0000-N-1481, nodalization was modified to separately model each of the four MSLs shown in Figure RAI-3b, "Modified Nodalization for a Single Steam Line." As a result, each sensitivity case included four RADTRAD models, one for each line with three well-mixed nodes per line. The response asserted the outboard steam line up to the turbine stop valve (TSV) at QCNPS, Units 1 and 2, is seismically qualified, so including holdup and deposition in this piping as part of the outboard compartment (third well-mixed node in Figure RAI-3b) conforms with the guidance in RG 1.183. The data used to calculate the steam line and condenser aerosol removal rates are provided in Tables RAI-3b and 3c and are duplicated from QDC-0000-N-1481 Sections 7.2 and 7.3. The sensitivity case results are summarized in Table RAI-3e: Sensitivity Study Results.

The NRC staff observed that the basis for the 2-micron AMMD particle size and the methodology for the 20-group particle size distribution was not fully described in the response to RAI No. ARCB-RAI-3. Further, in QDC-0000-N-1481, Revision 4, Section 4.6.9, it is stated, in part, that "for additional information on the models, data, and results refer to QDC-0000-N-2373 [Revision 0, "provided AST LOCA Aerosol Removal Factors and Margin Assessment" (Ref. 9.60)]" which was not with the RAI response. The NRC staff notes that neither the 2-micron AMMD particle size or the 20-group particle size distribution method were described in the QCNPS, Units 1 and 2, CLB for calculating design basis LOCA doses. As a result, the NRC staff conducted a regulatory audit from May 11 to 15, 2020, to gain a better understanding of the licensee's non-docketed information. The NRC staff's plan for the audit is located in ADAMS at Accession No. ML2019J983. By letter dated June 24, 2020 (ADAMS Accession No. ML20169A614), the NRC issued its summary of the regulatory audit.

The licensee assumed a 2-micron AMMD and GSD 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.

3.1.5 NRC Staff's Evaluation of Licensee's Sensitivity Analysis

The response to RAI No. ARCB-RAI-3, 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. The response stated that based on the sensitivity analysis the conservatism associated with crediting aerosol impaction on the first closed MSIV and aerosol deposition in the outboard steam lines was sufficient to offset the uncertainty introduced by the drywell spray effects on the aerosol deposition model.

As described in Section 3.1.4 of this SE, a total of seven sensitivity cases were performed from various combinations of breathing rate, MSIV impaction, and condenser holdup/aerosol deposition by varying the base case. As shown in Table RAI-3e: Sensitivity Study Results, the base case indicates that the conservative modelling of the drywell spray on the aerosol removal in the MSLs, without making other adjustments, results in increased doses. The NRC staff notes that the base case results indicate that while the calculated CR dose exceeds the 5 rem TEDE acceptance criterion, the off-site doses to members of the public remain within the dose acceptance criteria. The NRC staff also notes that the base case was produced for the purpose of conducting a sensitivity analysis and is not intended to replace the accident analysis of record. The accident analysis of record is the revised design basis LOCA radiological analysis as discussed in the letter dated March 31, 2020, in response to RAI No. ARCB-RAI-1 concerning obstructions in the drywell and RAI No. ARCB-RAI-2 concerning data and parameter discrepancies and the containment pathway added in the RADTRAD model. The accident analysis of record indicates that dose consequences comply with all the applicable dose acceptance criteria.

The March 31, 2020, letter asserted that the seven 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 LAR evaluation)

  As stated in the LAR, credit for full drywell spray lambdas (lambda represents removal rate per unit time) was not included in the licensee's evaluation. Because an evaluation and technical basis was not included in the sensitivity study, the NRC staff did not review the assertion that credit for full drywell spray lambdas is a conservatism in the AST LOCA model. Therefore, the NRC staff did 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 LAR evaluation)

  As stated in the LAR, credit for plateout and deposition in the drywell was not included in the evaluation. Because an evaluation and technical basis was not included in the sensitivity study, the NRC staff did not review the assertion that the proposed models' 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 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 main steam lines for holdup and deposition

  Although the leakage can exist in four MSLs simultaneously, it can also exist in three MSLs simultaneously. Because the dilution and deposition differ between MSLs, the assumed inputs and modeling of the allowed leakage can impact the postulated dose. RG 1.183 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 postulated dose." 10 CFR 50.67 states, in part, that "[t]he fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible." Based on the NRC staff's understanding, it is expected that modeling three MSLs rather than four MSLs would result in the most conservative postulated dose. Therefore, the NRC staff does not consider modeling all four MSLs for holdup and deposition to be a conservatism in the AST LOCA model.

- Outboard main steam line piping holdup and deposition

  Attachment 1 of the March 21, 2020, letter describing "Nodalization Changes" and in Table RAI-3e: Sensitivity Study Results, considered credit for holdup and deposition in the piping from the outboard MSIV up to the TSV. The RAI response stated, "The outboard steam line up to the turbine stop valve at QCNPS is seismically qualified, so including holdup and deposition in this piping as part of the outboard compartment (third well-mixed node in Figure RAI-3b) conforms with the requirements of RG 1.183." The response did not state that the TSV was seismically qualified; therefore, the NRC staff does not consider credit for holdup in the section of piping from the outboard MSIV to the TSV to be a conservatism in the AST LOCA model. With respect to the deposition in the sections of MSL between the outboard MSIVs and the TSV, this information was not provided as part of the accident analysis of record, and therefore the NRC did not review and evaluate this to be a conservatism in the AST LOCA model.

- More realistic CR operator breathing rate

  The LAR referenced breathing rate data from Table 6-17 of 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 CR dose consequence analysis. The March 21, 2020 letter, Attachment 1, Table RAI-3e: Sensitivity Study Results, referenced the RG 1.183 recommended breathing rate for the first 2 hours followed by reduced breathing rates from the EPA handbook of 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 response stated that the analysis considered the 95th percentile confidence level values from the EPA Handbook as light intensity work typical of a CR operator. As a result, the observed CR 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 Attachment 1, Table RAI-3e: Sensitivity Study Results, consideration of a more realistic CR breathing rate is observed to show a reduction (about 7%) on the CR dose (based upon the differences in dose results of

Sensitivity Cases S0 and S1). 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 design basis radiation exposures from “access to and occupancy” in 10 CFR 50.67 of the CR under accident conditions where CR operators would be expected to be at a higher level of stress and increased activities. Therefore, use of a more realistic CR breathing rate would not be considered in the NRC’s design basis determination of its acceptability.

- Aerosol impaction on the first closed MSIV

  Attachment 1, Table RAI-3e: Sensitivity Study Results, considered credit for aerosol impaction on the first closed MSIV in the RADTRAD model. As a result, the observed CR, EAB, and 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 Attachment 1, Table RAI-3e: Sensitivity Study Results, consideration of MSIV impaction is observed to show a reduction (up to about 36%) on the CR, EAB, and LPZ doses.

  The response referenced the NMP1 AST LOCA licensing basis described in H21C092 (ADAMS Accession No. ML070110240) that credits the phenomenon of impaction at the first closed MSIV. The response stated 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. NMP1 conservatively determined this impaction results in a Decontamination Factor (DF) of 2, which is modeled as a 50% filter in the transfer pathway through the first closed MSIV. This reduction is only accounted for once in each MSL. The response stated that this approach was previously approved for NMP1 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 NMP1 included the following assumption in their MSIV leakage dose consequence analysis:

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

  The NRC staff’s SE associated with NMP1 dated December 19, 2007, 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 Unit 1] NMP1. The contribution of this impaction DF to the overall iodine activity decontamination, does not lead to an excessive overall credit for iodine removal in the MSLs. Based on the approximate DF of 4 that credits for removal by sedimentation (See Section 3.2.1.2.1.4), combined with this DF of 2, the licensee is assuming less than a 90% overall iodine removal efficiency in the steam lines. If this MSIV leakage pathway were modeled using a well-mixed model, as described and previously approved in AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," December 9, 1998, the calculated activity removal in the MSLs would be analogous to that calculated by the licensee. Therefore, the NRC staff finds the overall iodine removal credited by to be acceptable, as modeled for NMP1.

The NRC staff notes that the above excerpt from the NRC staff's NMP1 SE clearly states that it does not generally endorse taking credit for impaction when modeling removal of particulates in MSLs 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 NMP1 was acceptable. Under 10 CFR 50.40, the NRC staff makes determinations 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 MSLs following a LOCA. Therefore, the NRC staff does not consider credit for MSIV impaction to be a conservatism in the AST LOCA model.

- Condenser holdup and deposition

  The LAR stated a further conservatism that is not currently modeled in QDC-0000-N-1481 is the holdup and aerosol deposition provided by the condenser. The LAR stated 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 LAR stated that the sensitivity analysis modelled an MSIV leakage pathway to the condenser through the drain lines from the MSL piping between the MSIVs. The LAR stated that this model neglects any holdup and deposition in the outboard MSL piping and that modelling the release to the condenser from the piping between the MSIV is consistent with other plants in the Exelon fleet (e.g., LaSalle and Limerick). The LAR stated that operating experience associated with the North Anna earthquake and post-Fukushima Dai-ichi 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 LAR concluded that it is reasonable to assume that the condenser pathway could be made available for mitigating the consequences of MSIV leakage. The LAR stated the data used to calculate the steam line and condenser aerosol removal rates were provided in Tables RAI-3b and 3c are from QDC-0000-N-1481 Sections 7.2 and 7.3.

  Attachment 1, Table RAI-3e: Sensitivity Study Results, stated that condenser credit is considered in the RADTRAD model. As a result, the observed CR, EAB, and LPZ doses are effectively reduced when compared to the Base Sensitivity Case. The LAR stated

that condenser credit has the capability to ensure post-LOCA releases remain well within the 10CFR 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 Attachment 1, Table RAI-3e: Sensitivity Study Results, consideration of condenser credit is observed to show a reduction (up to about 96%) on the CR, EAB, and LPZ doses.

The sensitivity results demonstrate that the condenser is very effective in substantially reducing the dose consequences from MSIV leakage. While other elements assessed in the licensee's sensitivity analysis provided relatively small decreases in the calculated doses, the evaluation of the condenser's mitigation properties provided a substantial dose reduction. The NRC staff notes that while RG 1.183 recommends for design basis LOCA radiological analysis that the structures, systems, and components (SSCs) credited with creating a pathway to the condenser 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 3.1.7 of this SE.

In addition, the probability of an event resulting in substantial fuel melt such as a LOCA followed by a significant unrelated seismic event is very low. Specifically, RG 1.183, Appendix A, describes assumptions for evaluating the radiological consequences of a LOCA. Section 6 of Appendix A describes assumptions on MSIV 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 dose acceptance for the LOCA analysis described in RG 1.183 is based on the regulatory acceptance criteria for what is commonly termed the maximum hypothetical accident (MHA) or the maximum credible accident. The MHA is that an accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite dose

consequences evaluated based on design-basis containment leakage.

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

The LAR also stated there are other significant conservatisms associated with the AST LOCA model. Specifically, CR atmospheric dispersion (χ/Q) factors (values) have readily defined uncertainty distributions and if incorporated would demonstrate there is a substantial amount of margin in the input parameters. The LAR 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% confidence level ensuring there is reasonable assurance that the acceptance criteria will not be exceeded. Therefore, the NRC staff does not endorse the concept of including χ/Q values as an element to be considered in dose consequence sensitivity analyses.

3.1.6 Transport of Radioactivity in the Drywell

RG 1.183, Appendix A, Section 3.1, states, in part:

> The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs [pressurized-water reactors] or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell.

RG 1.183, Appendix A, Section 3.3, states, in part, that the “Evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops.” RG 1.183, Appendix A, Section 6.1, states, in part, that the “activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage.”

Enclosure B, Section 2.1.2, “Transport in Primary Containment,” page 9 of the LAR states, in part, that “For calculating the MSIV leakage flow rates between the drywell and the environment, the flow rate analysis is based on the total drywell volume during the first 2 hours of the LOCA, and then the combined drywell plus suppression chamber air volume after 2 hours, at which time the containment volume is expected to become well mixed following the restoration of core cooling.”

Section 7.2.3, “MSIV Leakage During 2-24 hrs,” page 51 of the LAR, states, in part:

> Two hours after a LOCA, the drywell and suppression chamber volumes are expected to reach an equilibrium condition and the post-LOCA activity is expected to be homogeneously distributed between these volumes. The homogeneous mixing in the primary containment will decrease the activity concentration and therefore decrease the activity release rate through the MSIVs. To model the effect of this mixing, the MSIV flow rate used in the RADTRAD model is decreased by calculating a new leak rate based on the combined volumes of the drywell and suppression chamber.

Enclosure B, Section 2.1.2, “Transport in Primary Containment,” page 9 of the LAR references Table 2 of AEB 98-03, which shows the dependence of radiological consequences on containment mixing conditions for the Perry Nuclear Power Plant. However, the Perry Nuclear Power Plant has a Mark III containment, which is significantly different than the Mark I containment at QCNPS. These differences are not addressed in the LAR.

The LAR proposes a significant change to the CLB transport modeling in primary containment by adding a compartment in the drywell to credit sprays and by crediting transport between the sprayed and unsprayed portions of the drywell. As a result, it is not clear that the assumption of equilibrium conditions at 2 hours exists between drywell and wetwell volumes. The proposed credit for sprays and the addition of the sprayed compartment decreases the activity in the drywell from the activity in the CLB and, therefore, will create a difference in the modelled activity in the sprayed drywell compartment as compared to the activity in the wetwell.

From the NRC staff’s review of Enclosure B, Attachment 13.1 - RADTRAD Output File “QDC39CL02.o0,” starting on page 404 of the LAR, it appears that the I-131 activity concentrations for the sprayed and unsprayed portions of the drywell do not reach equilibrium conditions until after 5 hours beyond the time when RHR drywell sprays are assumed to terminate at 4 hours post-accident for aerosol removal.

To address the concern on the transport of radioactivity in the drywell, the NRC staff issued RAI No. ARCB-RAI-4 requesting an explanation of why the high flow rates necessary to create equilibrium conditions between the drywell and wetwell would exist for the time period from 2 hours in the QCNPS, Units 1 and 2, design.

The letter dated March 31, 2020, responded to RAI No. ARCB-RAI-4 and explained the assumption of equilibrium conditions between the drywell and wetwell is based on the steaming/condensing phenomenon associated with the restoration of core cooling at 2 hours. The response explained that although the wetwell is not modeled separately in the containment leakage and MSIV leakage RADTRAD cases, the wetwell volume is used in the MSL flow rate calculations starting at 2 hours, and crediting drywell sprays for airborne fission product removal does not change the well-mixed assumption.

The response stated that the RADTRAD modeling is based on separating the unsprayed and sprayed drywell volumes because the drywell sprays are assumed to cover less than 90% of the drywell volume. The response explained that the RADTRAD modeling is intended to conservatively concentrate the airborne activity in the sprayed volume directly connected to the MSIV leakage pathways and to maximize dose, not to accurately reflect the thermal-hydraulic conditions that would be present in the drywell. The response further explained the discrepancy noted in the (iodine) I-131 inventory between the sprayed and unsprayed volumes is unrelated

to the well mixed assumption between the drywell and wetwell at 2 hours and is a byproduct of the conservative modeling inside the drywell.

RG 1.183, Appendix A, Section 3.3, states, in part, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown." Based on a review of the response to RAI No. ACRB-RAI-4, the NRC staff finds the explanation reasonable because the sprays are assumed to cover less than 90% of the dry well volume in the RADTRAD modeling and high flow rates are expected to be present in the drywell from thermo-hydraulic conditions following a design basis LOCA and, therefore, is acceptable.

3.1.7 NRC Staff Risk and Engineering Insights

The LAR was not submitted as a formal "risk informed" submittal with probabilistic risk assessment information in accordance with the guidance of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." (ADAMS Accession No. ML17317A256). Thus, the NRC staff's findings are primarily based on traditional deterministic review approaches.

In the Staff Requirements Memorandum (SRM) to SECY-19-0036 (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 and the Office of Nuclear Reactor Regulation's (NRR) efforts to advance the reactor safety program towards becoming a more modern and risk-informed regulator and the NRC's Principles of Good Regulation. Since the application is not a fully risk-informed submittal (with probabilistic risk information), the staff does not apply risk as the basis for acceptance of a request; however, the following risk and engineering insights inform the technical review by supporting the deterministic safety conclusions and enhance the technical reviewers' confidence in their technical evaluations of reasonable assurance.

As described in Section 3.1.5 of this SE, the LAR stated that aerosol holdup and deposition provided by the condenser is not modeled in QDC-0000-N-1481 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 LAR 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 NRC staff evaluated the seismic capacity of the systems, structure and components (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 probabilistic and risk information, to complete the evaluation. The NRC staff also leveraged more recent relevant operating experience, such as that obtained from the Fukushima Dai-ichi 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 it being unavailable to serve as a volume for holdup and retention is very low.

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

Based on the available information and assessments, using conservatively biased assumptions, about the seismic capacity of the SSCs in the realistic pathway, the NRC staff determined that there is high confidence that the main steam lines 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 design basis accident 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 reasonable assurance finding based on its deterministic review.

3.1.8 Dose Consequence Conclusion

As described above, the NRC staff reviewed the dose consequences used to assess the radiological impacts of the proposed license amendment at QCNPS, Units 1 and 2. The NRC staff finds the analysis methods and assumptions consistent with the regulatory requirements and guidance specified in Section 2.2 of this SE. The NRC staff concludes, with reasonable assurance, based in part on the risk insights provided in Section 3.1.7 of this SE 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 CR doses will comply with the cited acceptance criteria. The NRC staff further finds reasonable assurance that QCNPS, Units 1 and 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 Environmental Qualification (EQ)

The proposed LAR would increase the MSIV leakage rate for all four MSLs and revise the leakage rate through each MSIV leakage path in TS SR 3.6.1.3.10. The NRC staff evaluated whether equipment and components would remain bounded by the existing EQ due to the proposed change.

The LAR stated that the EQ doses are not impacted by the proposed change because the current EQ design basis does not include source term in the MSLs downstream of the MSIVs. However, the LAR did not provide an evaluation of the impact of the MSIV increased leakage rate on temperature, pressure, or humidity on electrical equipment. Therefore, the NRC staff issued a RAI requesting demonstration that the temperatures, pressures, and humidity remain bounded by the existing EQ for electrical equipment impacted by the MSIV increased leakage rate. The letter dated February 24, 2020, provided a response to the subject RAI and explained that the bounding accident temperature and pressure profiles in the main steam tunnel and turbine building are associated with a high energy line break (HELB) in the steam tunnel. When the increased MSIV leakage is considered, the HELB temperature and pressure profile in these zones continues to bound the LOCA profile. Additionally, the accident humidity in these zones

is already assumed to be 100%. Therefore, the proposed increase in allowable MSIV leakage would contribute no additional environmental impact to equipment qualified for use in the main steam tunnel or the turbine building. Based on its review of the RAI response, the NRC staff finds that the LAR, as supplemented, adequately showed that the temperature, pressure, and humidity remain bounded by the existing EQ for electrical equipment as a result of the proposed change.

Also, from the review of the LAR, it was unclear to the NRC staff whether the impact of the proposed change was considered on non-safety related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. Therefore, the NRC staff issued a RAI requesting a description of how the impact of the proposed change was considered on non-safety related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. The letter dated February 24, 2020, in response to the RAI stated that since there is no change to EQ design basis temperatures, pressure, humidity, or radiation values, the proposed increase in MSIV leakage has no impact on non-safety related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment. Based on the response, the NRC staff finds that the LAR, as supplemented, has provided reasonable assurance that the proposed change will not adversely affect the potential for non-safety related electrical equipment to prevent satisfactory accomplishment of safety functions.

The NRC staff further requested confirmation of whether any components are being added to the EQ equipment list to comply with 10 CFR 50.49 due to the proposed change. The letter dated February 24, 2020 stated that there are no components that are being added to the EQ equipment list due to the proposed increase in allowable MSIV leakage. Based on the LAR, as supplemented, the NRC staff finds that no new electrical equipment needs to be added to the QCNPS, Units 1 and 2, 10 CFR 50.49, EQ program, as a result of the proposed change to the MSIV leakage rate.

3.2.1 EQ Conclusion

Based on its review of the information in the LAR, as supplemented, the NRC staff finds that the proposed changes will have no adverse impact on the QCNPS EQ program or its ability to continue to meet the requirements of 10 CFR 50.49.

3.3 TSs Changes

3.3.1 MSIV Leakage Rate Limits

The postulated LOCA radiological consequences analysis is based on leakage (see Section 3.1.2 above) that is less than a total MSIV leakage of 250 scfh at 43.9 pounds per square inch gauge (psig) for QCNPS, Unit 1, and 350 scfh at 43.9 psig for QCNPS, Unit 2, and was assumed to occur as follows:

1) 100 scfh for QCNPS, Unit 1, and 125 scfh for QCNPS, Unit 2, through the steam line with the "failed" MSIV.

2) 100 scfh through first intact steam line for QCNPS, Unit 1, and 125 scfh for QCNPS, Unit 2.

3) 50 scfh through the second intact steam line for QCNPS, Unit 1, and 100 scfh for QCNPS, Unit 2.

4) 0 scfh through the fourth steam line for both units.

SR 3.6.1.3.10 leakage rate through each MSIV leakage path when tested at ≥ 25 psig is increased for each unit from ≤ 34 scfh to ≤ 62.4 scfh for QCNPS, Unit 1, and ≤ 78 scfh for QCNPS, Unit 2, and the combined leakage rate for all MSIV leakage paths when tested at ≥ 25 psig is increased for each unit from ≤ 86 scfh to ≤ 156 scfh for QCNPS, Unit 1, and ≤ 218 scfh for QCNPS, Unit 2, in the LAR. This SR ensures that MSIV leakage is properly accounted for and remains consistent with the postulated LOCA radiological consequences analysis.

The proposed TSs MSIV leakage rate limits are less than those in the revised MSIV leakage limits used in the reanalysis of the postulated LOCA radiological consequences which is described in Enclosure B of the LAR, as supplemented. The proposed change to the TSs MSIV leakage rate limits are consistent with the reanalysis of the postulated LOCA radiological consequences and therefore will require action before the analysis limit is reached. All other aspects of this SR are unchanged and the bases for those requirements are unchanged and therefore were not evaluated by the NRC staff. The NRC staff finds there is reasonable assurance that: (1) 10 CFR 50.36(b) will continue to be met, and (2) 10 CFR 50.36(c)(3) will continue to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation are met. Therefore, the proposed revision to SR 3.6.1.3.10 is acceptable.

3.3.2 Addition of TS 3.6.2.6, "Residual Heat Removal (RHR) Drywell Spray"

Evaluation of new TS LCO 3.6.2.6

The postulated LOCA radiological consequences analysis assumes, a minimum of one RHR drywell spray subsystem using one RHR pump is required to adequately scrub the inorganic iodines and particulates from the primary containment atmosphere. To ensure that these requirements are met, two RHR drywell spray subsystems must be operable with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is operable assuming the worst case single active failure. New TS 3.6.2.6 requires two RHR drywell spray subsystems to be operable in Modes 1, 2, and 3. In Modes 1, 2, and 3 (i.e., power operation, startup, and hot shutdown, respectively), there is considerable energy in the reactor core and a LOCA could release fission products into the primary containment. In Modes 4 and 5 (i.e., cold shutdown and refueling, respectively), there is less energy in the reactor core such that the probability and consequences of a LOCA is reduced due to the reduced pressures and temperatures in these modes. The NRC staff finds that maintaining RHR drywell spray subsystems operable is not required in Mode 4 and 5 and that the proposed applicability is acceptable because there is less energy in the reactor core such that the probability and consequences of a LOCA is reduced due to the reduced pressures and temperatures in these modes. In addition, the applicability of Mode 1, 2, and 3 is consistent with the applicability of the emergency core cooling system TS which also functions to limit the release of radioactive materials to the environment following a LOCA.

Evaluation of Action A

QCNPS, Units 1 and 2, TSs define Actions as that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.

New TS 3.6.2.6, Condition A, is entered when one RHR drywell spray subsystem becomes inoperable and Required Action A.1 requires restoring the inoperable RHR drywell spray subsystem to operable status within 7 days. The NRC staff finds Action A, is acceptable because the remaining operable RHR drywell spray subsystem is adequate to perform the primary containment fission product scrubbing function and there is a low probability of a LOCA occurring during the 7-day period.

Evaluation of Action B

New TS 3.6.2.6 Condition B is entered when two RHR drywell spray subsystems become inoperable and Required Action B.1 requires restoring one RHR drywell spray subsystem to operable status within 8 hours. The NRC staff finds Action B is acceptable because of the low probability of a LOCA occurring during the short completion time.

Evaluation of Action C

New TS 3.6.2.6, Condition C, is entered when operators are unable to restore the RHR drywell spray subsystems to operable status within the completion time under Condition A or B. Required Action C.1 requires the unit to be in Mode 3 in 12 hours followed by Mode 4 in 36 hours, as required by Required Action C.2. The NRC staff finds Actions C is acceptable because the condition requires the operators to place the unit in a condition in which the LCO no longer applies. In addition, the proposed completion times allow a reasonable amount of time to reach Mode 3, hot shutdown and Mode 4, cold shutdown in an orderly manner and without challenging plant systems from full power operation in Mode 1.

Evaluation of SRs

New SR 3.6.2.6.1 requires verification that each RHR drywell spray subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position. New SR 3.6.2.6.2 requires verification that each drywell spray nozzle is unobstructed. New SR 3.6.2.6.3 requires verification that the RHR drywell spray subsystem locations susceptible to gas accumulation are sufficiently filled with water.

The NRC staff finds the new SRs are acceptable because: (1) SR 3.6.2.6.1 ensures a flow path exists between the RHR pumps and the drywell spray nozzles so that the flow path is available following a LOCA; (2) SR 3.6.2.6.2 ensures there are no blockages that would affect the spray pattern which would invalidate the input and assumptions used the QCNPS reanalysis of the postulated LOCA radiological consequences in the LAR, as supplemented; (3) SR 3.6.2.6.3 ensures the normally water-filled lines of the RHR system do not have gas accumulation which is necessary for proper operation of the RHR drywell spray subsystems and may also prevent water hammer and pump cavitation; and (4) existing SR 3.6.2.3.2 verifies each required RHR pump develops a flow rate greater than or equal to 5000 gpm through the associated heat exchanger while operating in the suppression pool cooling mode, which is substantially greater than the 2,352 gpm assumed in the reanalysis of the postulated LOCA radiological consequences.

The frequency of the new SRs is in accordance with the Surveillance Frequency Control Program. The May 23, 2019 supplement to the LAR stated that:

> The surveillance frequencies that presently exist in TRM [Technical Requirements Manual] 3.6.a will be duplicated in the Surveillance Frequency Control Program (SFCP) at the same time the proposed amendment, if approved, is implemented at the site … and the associated SR frequencies will be added to TRM Appendix I, which is the QCNPS SFCP list of controlled frequencies … SR 3.6.2.6.1 frequency for valve positions will remain 31 days … SR 3.6.2.6.2 frequency for the Spray Nozzles will remain 10 years … The new SR 3.6.2.6.3 checking for gas accumulation will match the corresponding RHR suppression pool spray SR frequency of 184 days … The TRM frequencies will not change when the requirements are moved back to the TS. The due dates for the existing surveillances will determine the next applicable due date for SR 3.6.2.6.1 (valve position) and SR 3.6.2.6.2 (spray nozzles). SR 3.6.2.6.2 surveillances are scheduled for completion within the current grace period which ends on 11/10/2021 for Unit 1 and 9/16/2020 for Unit 2. The gas accumulation surveillance, SR 3.6.2.6.3, is new for drywell spray, but it will be bundled into the existing RHR related surveillance for gas accumulation, and thus be tied to the existing due dates…
>
> Motor-operated valves 1(2)-1001-23A(B) and 1(2)-1001-26A(B), A(B)Containment Spray Loop Upstream and Downstream Stop Valves respectively, are explicitly modeled in the Quad Cities Full Power Internal Events Level 1 and Level 2 PRA [Probabilistic Risk Analysis] fault trees and the Quad Cities Fire PRA fault trees, capturing the drywell spray function for primary containment cooling. The drywell spray function for fission product scrubbing is implicitly modeled, where successful spray implies successful scrubbing. The drywell spray nozzles are not explicitly modeled in the PRA. However, the proposed Surveillance Requirement test intervals for Residual Heat Removal Drywell Spray (31 days for SR 3.6.2.6.1, 10 years for SR 3.6.2.6.2, and 184 days for SR 3.6.2.6.3) can be represented in the Quad Cities PRA either explicitly or implicitly.

The new SRs 3.6.2.6.1, 3.6.2.6.2, and 3.6.2.6.3: (1) do not reference other approved programs for the specific interval, (2) are not purely event driven, (3) are not event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs, (4) are not related to specific conditions or conditions for the performance of a SR, and (5) can be modeled either directly or implicitly in the plant specific probabilistic risk analysis (PRA). The NRC staff finds the base surveillance frequency interval for SR 3.6.2.6.1 is acceptable based on operating experience because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. The base surveillance frequency interval for SR 3.6.2.6.2 is acceptable based on operating experience because it is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state. The base surveillance frequency interval for SR 3.6.2.6.3 is acceptable because it takes into consideration the gradual nature of gas accumulation in the RHR drywell spray system piping and the procedural controls governing system operation.

3.3.3 TS 3.6.4.1, ‘‘Secondary Containment’’

LAR Enclosure A, QDC-7500-M-2341, Revision 0, “Quad Cities Units 1 & 2 Secondary Containment Drawdown Analysis,” calculated that a single train of SGT would draw secondary containment pressure down from the normal operating minimum secondary containment vacuum of 0.1 inch water gauge to the required minimum post-LOCA vacuum of 0.25 inches water gauge in 22.9 minutes in the bounding scenario evaluated. The AST analysis assumed the required secondary containment vacuum would be reached in 25 minutes. LCO 3.6.4.1 “The secondary containment shall be OPERABLE” required action A.1 to restore operability has a completion time of 4 hours. The proposed note would allow the LCO not to be met for up to 4 hours if an analysis demonstrated that one SGT subsystem could establish the required secondary containment vacuum. During normal operation, conditions may occur that result in SR 3.6.4.1.1 not being met for short durations. Occasionally wind gusts may lower external pressure on secondary containment walls or normal ventilation system component failures or realignments may affect secondary containment vacuum. These conditions are most often not degradations of the secondary containment boundary or the SGT system ability to accomplish its specified safety function.

The note permits confirmation of secondary containment operability by confirming that one SGT subsystem could accomplish the required drawdown. This confirmation is necessary to apply the limited exception to meeting the SR acceptance criterion. While the duration of these occurrences is anticipated to be brief and infrequent, the allowance is permitted for a maximum of 4 hours, which is consistent with the time permitted for secondary containment to be inoperable per Condition A of LCO 3.6.4.1. The NRC staff finds that addition of this note to SR 3.6.4.1.1 is acceptable since any additional risk of not restoring the vacuum pressure is minimal and the dose limits associated with 10 CFR 50.67 will be met.

3.4 Credited Operator Actions

3.4.1 Description of Credited Operator Action

Section 2.1.2, “New TS 3.6.2.6 Residual Heat Removal (RHR) Drywell Spray,” of the LAR indicates that operator actions associated with the use of drywell sprays are credited to reduce airborne activity in the drywell. The drywell sprays are expected to mitigate the consequences of a postulated LOCA by scrubbing radionuclides from the air in the drywell.

The CLB does not credit operator actions to initiate the drywell sprays for the purpose of scrubbing radionuclides from the air, however, it does credit these same operator actions as a means of removing heat from the drywell. This is described in Section 7.4.1 of the UFSAR which states: “Containment cooling mode of RHR is initiated manually from the control room by alignment of the proper combination of valves, pumps, and heat exchangers. No automatic start function is provided.”

The LAR subsection entitled “Drywell Spray” found in Section 3.1, “Evaluation of Proposed Change to the MSIV Leakage Rate Limits,” describes crediting of the Residual Heat Removal (RHR) system for both functions. Section 3.1 of the LAR also indicates that the manual initiation of dry well sprays is tracked in the Operator Response Time Program.

3.4.2 Evaluation of Credited Operator Actions

The newly credited operator actions are the same as operator actions credited in the CLB. The only difference is the reason for performing these actions (removing heat as currently described in the UFSAR versus scrubbing radionuclides as described in the LAR). Therefore, it is reasonable to assume that any human factors analyses used to validate these actions should remain valid.

The operator actions identified in the LAR are monitored by the operator response time program. This ensures that the actions remain feasible over time.

3.4.3 Operator Actions Conclusion

The considerations described above are consistent with the state-of-the-art human factors principles described in NUREG-0711 and NUREG-1764 and, therefore, the NRC staff finds this treatment reasonable to meet 10 CFR 50.34(f)(2)(iii) and therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments on June 2, 2020. The Illinois State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and the surveillance requirements. 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (84 FR 45543). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

amendment will not be inimical to the common defense and security or to the health and safety of the public.

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SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 281 AND 277 TO INCREASE ALLOWABLE MAIN STEAM ISOLATION LEAKAGE (EPID L-2019-LLA-0045) DATED JUNE 26, 2020

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DATE	9/4/2019	6/1/2020	10/9/2020	6/1/2020
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