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

January 3, 2019

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

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT 221 REGARDING REVISED ALTERNATIVE SOURCE TERM DOSE CALCULATION (EPID L-2018-LLA-0004)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 221 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated January 9, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18009B037), as supplemented by letter dated August 29, 2018 (ADAMS Accession No. ML18242A156).

The amendment revises the loss-of-coolant accident dose calculation in the Clinton Power Station, Unit 1, updated safety analysis report and revises the technical specifications acceptance criteria for primary containment feedwater penetration leakage.

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

Sincerely,

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures:
1. Amendment No. 221 to NPF-62
2. Safety Evaluation

Cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 221
License No. NPF-62

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 January 9, 2018, as supplemented by letter dated August 29, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-62 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 221, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 3, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 221

CLINTON POWER STATION, UNIT NO. 1

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Facility Operating License and 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
License NPF-62 Page 3	License NPF-62 Page 3
TSs 3.6-19a	TSs 3.6-19a

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

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

(6) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation; and

(7) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3473 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

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

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 221, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	----------------NOTE---------------- Only required to be met in MODES 1, 2, and 3. ------------------------------------ Verify that the combined leakage rate for both primary containment feedwater penetrations is $\leq$ 1.5 gpm when pressurized to $\geq$ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.
SR 3.6.1.3.12	Verify each instrumentation line excess flow check primary containment isolation valve actuates within the required range.	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. 221 TO FACILITY OPERATING LICENSE NO. NPF-62

EXELON GENERATION COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter dated January 9, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18009B037), as supplemented by letter dated August 29, 2018 (ADAMS Accession No. ML18242A156), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request (LAR) to modify the Clinton Power Station (CPS), Unit 1, alternative source term (AST) dose consequence analysis for the loss-of-coolant accident (LOCA) and revise the CPS technical specifications (TSs) regarding the acceptance criteria for feedwater penetration leakage. The revised CPS LOCA control room (CR) dose consequence remains within the Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67 requirement; however, the increase in consequences is more than minimal, and as such, requires prior U.S. Nuclear Regulatory Commission (NRC) approval, in accordance with 10 CFR 50.59(c)(1)(iii). The August 29, 2018, supplement, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The CR heating, ventilating, and air conditioning (HVAC) system is an engineered safety feature system and is designed with redundancy to ensure operation and habitability under normal and accident conditions. The system is designed to ensure that main CR personnel can remain inside the spaces served by the system, in compliance with General Design Criteria (GDC) 19 of Appendix A to 10 CFR.

The CR HVAC system is comprised of two full-capacity, redundant equipment trains. Each train has a 100 percent capacity recirculation (standby) filter train consisting of high-efficiency air filter, an absorber (charcoal filter) for fume and odor removal (normally bypassed), a humidification system, a supply air fan, and a blow-through type air-handling unit comprised of cooling coil, heating coil, and zone mixing dampers. Individual zone ducts from each train are cross-connected to common ducts and supply air to the corresponding zone. Return air ducts from the areas served by the CR HVAC system are connected to two 100 percent capacity redundant return air fans discharging into their respective mixing plenum upstream of the recirculation supply air filters on each air-handling equipment train. During normal operations, flow from the return air fans bypass the standby recirculation air filters. Dampers provided in the ductwork have the capability to close the bypass and direct the flow the through the recirculation filter train. The system also includes an exhaust fan serving the locker room area.

The outside air is normally brought in through one of two minimum air intakes and ducts connected to the operating return air fan suction. These two intakes are physically separated by over 375 feet (ft), and are called "minimum" outside air intakes since one of them is used to supply the minimum required makeup air during normal and abnormal conditions. The two minimum air intake paths to the suction of the return air fans also have the capability by means of closing and opening appropriate dampers to direct the minimum intake flow through one of two 100 percent makeup filter trains to the return air fan suction, when needed. During normal conditions intake flow to the return fans is unfiltered but does pass through normal filters in the air handling unit after mixing with the recirculation flow. The minimum quantity of outside air provides makeup air for expected leakages (delineated in updated safety analysis report (USAR), Table 6.4-1 (ADAMS Accession No. ML16306A075)), and locker room exhaust fan operation and still maintain not less than 0.125 inch water (H_20) gage positive pressure in the CR envelope with respect to the adjacent areas under all station operating conditions with the exception of certain habitability modes of operation. During chlorine mode, all intakes are closed and the system runs in recirculation mode. During normal operating conditions, the makeup flow rate is approximately 4000 cubic feet per minute (cfm), 1000 cfm of which is exhausted to atmosphere by the locker room exhaust fan. Maintaining positive pressure precludes infiltration of unconditioned air during all modes except when the system is in complete recirculation mode with the minimum outside intake dampers closed. Maintaining positive pressure during normal mode is not a safety function.

High radiation measured at either minimum outside air intake automatically closes the normal intake dampers and initiates operation of one of two 100 percent standby makeup air filter trains, which in turn, opens the appropriate standby makeup filter train inlet and outlet dampers, depending on which CR HVAC system train is operating. This in turn sends a signal to close and open appropriate dampers to divert flow from the return air fans through the recirculation air filter trains and on to the supply fan and air-handling unit. For the removal of radioactive contaminants, minimum outside air is thus introduced through a demister, electric heater, medium filter, high-efficiency particulate air (HEPA) filter, iodine adsorbing beds, and downstream HEPA filter in the makeup filter train. The pre-filter limits large particulate loading of the HEPA filter, and the single-stage electric heater assures no higher than 70 percent relative humidity air entering the charcoal. The makeup air filter trains are capable of removing 99.95 percent of all particulate matter larger than 0.3 microns and no less than 99 percent of all forms of iodine. The recirculation charcoal filter trains are capable of removing no less than 70 percent of all forms of iodine. The high radiation signal also closes the locker room exhaust dampers which trips the exhaust fan.

The licensee stated that the LOCA analysis supporting the AST implementation assumed credit for dual, remote CR HVAC minimum air intakes. By letter dated September 1, 2005 (ADAMS Accession No. ML052570461), the NRC issued Amendment 167 to NPF-62 approving the AST implementation at CPS. The AST dose calculations were revised twice since the initial AST implementation. In May 2015, a design change to revise the operating cycle to the current 12 months was implemented in accordance with the requirements of 10 CFR 50.59, based on the determination that the resultant change to the dose values is not more than minimal. An additional modification to the LOCA dose analysis was submitted to the NRC on January 29, 2016, to revise the post-LOCA drawdown time for secondary containment from 12 minutes to 19 minutes. The NRC approved this change by Amendment 210 to NPF-62 (ADAMS Accession No. ML16217A332).

The change requested in the licensee's January 9, 2018, letter, removes a reduction factor credit for dual remote CR outside air intakes that had been previously misapplied. Pursuant to Regulatory Position 3.3.2.3 in Regulatory Guide (RG) 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (ADAMS Accession No. ML031530505), if a dual ventilation system design allows the operator to manually select the least contaminated outside air intake as a source of outside air makeup and close the other intake, the atmospheric dispersion factors (χ/Q values) for the favorable intake may be reduced by a factor of four to account for the dual inlet and the expectation that the operator will make the proper intake selection. This protocol should be used only if the dual intakes are in different wind direction windows and if there are redundant, engineered safety-feature-grade radiation monitors within each intake with CR indication and alarm to monitor the intakes. However, EGC has subsequently determined that the factor of four reduction in χ/Q values is no longer applicable due to single-failure issues with the outside air intake dampers. Following a loss of divisional power (i.e., a single failure), the ability to select the more favorable remote intake from a CR radiological dose perspective would not be possible.

Consequently, the current credit taken for a dual CR remote intake can no longer be taken due to single-failure issues with the current system design. Based on this, the current reduction in the atmospheric dispersions by a factor of four to account for selection of the most favorable remote intake is no longer applicable. This change results in an increase in the CR χ/Q values based on the elimination of a factor of four reduction factor.

2.0 REGULATORY EVALUATION

Amendment No. 167 to Facility Operating License No. NPF-62 for the CPS, Unit 1, dated September 19, 2005 (Accession No. ML052570461), approved the application of an AST methodology. This LAR contains the NRC staff's safety evaluation (SE) of the licensee's radiological analysis assumptions and methods used to support the adoption of the AST methodology.

Amendment No. 210 to Facility Operating License No. NPF-62 for the CPS, Unit 1 dated August 17, 2016 (Accession No. ML16217A332), approved the application of a revised AST. This license amendment contains the NRC staff's SE of the licensee's LOCA radiological analysis incorporating an increase in the secondary containment drawdown time from12 minutes to 19 minutes.

The NRC staff's evaluation of the proposed atmospheric dispersion factors is based upon the following regulations, regulatory guides, and guidance documents:

- 10 CFR 50.67, "Accident source term," states that: (1) 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 0.25 sievert (Sv) (25 roentgen equivalent man (rem)) total effective dose equivalent (TEDE); (2) 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 during the entire period of its passage would not receive a total radiation dose in excess 0.25 Sv (25 rem) TEDE; and (3) adequate radiation protection is provided to permit access to and occupancy of the CR under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

- 10 CFR 50.36, "Technical Specifications" establish the regulatory requirements related to the contents of the TSs. Pursuant to 10 CFR 50.36(c), TSs are required to include items in the following specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

- Section 50.36(c)(3) of 10 CFR states that SRs 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.

- 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the nuclear reactor safety under normal conditions and to maintain the reactor in a safe condition under accident conditions, including LOCA. Adequate radiation protection is to be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of specified values.

- NUREG–0800, Standard Review Plan (SRP) Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Revision 3, dated March 2007 (ADAMS Accession No. ML063600394) relates to atmospheric dispersion factors used for the assessment of consequences related to atmospheric radioactive releases to the control room

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, dated March 2007 (ADAMS Accession No. ML070350028), includes guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing and operation

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, dated February 1983 (ADAMS Accession No. ML003740205), provides guidance on appropriate dispersion models for estimating offsite relative air concentrations (χ/Q values) as a function of downwind direction and distance (i.e., at the exclusion area boundary (EAB) and outer boundary of the low population zone (LPZ)) for various short-term time periods (up to 30 days) after an accident

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792). This RG provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This RG establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This RG also identifies acceptable radiological analysis assumptions for use in conjunction with an acceptable AST.

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," dated June 2003, which discusses acceptable approaches for estimating short-term (i.e., 2 hours to 30 days after an accident) average χ/Q values near the buildings at control room ventilation air intakes and at other locations of significant air in-leakage to the control room envelope due to postulated Design-basis Accident (DBA) radiological airborne releases

3.0 TECHNICAL EVALUATION

The current licensing basis (CLB) dose consequence LOCA takes advantage of the CR duel remote air intakes by crediting the ability of the CR operators to select the remote intake with the lowest radioactivity. The design basis dose consequence analysis assumes the worst-case single failure, that is, the loss of one of the safety-related power busses referred to as a loss of divisional power. The current configuration of the CR duel intakes precludes the ability to select the more favorable remote intake with a loss of divisional power. Consequently, the previous credit taken for dual remote CR intakes can no longer be taken due to single-failure issues with the current system design. To address this issue, the licensee has submitted an LAR to revise the CLB dose consequence LOCA accident analysis eliminating the credit for the duel remote CR intake system.

The licensee's revised LOCA dose consequence analysis includes the following changes to the CLB analysis:

1. A revision to the CR atmospheric dispersion values used in the LOCA analysis;
2. A reduction in the feedwater isolation valve liquid leakage from 2.0 gallons per minute (gpm) to 1.5 gpm;
3. A reduction in the feedwater isolation valve air leakage from 10.98 cfm to 8.64 cfm;
4. A reduction in the CR inleakage from 1,100 cfm to 1,000 cfm;
5. Taking credit for mixing in 50 percent of the secondary containment volume; and,
6. Incorporating a new core inventory based on a sensitivity study to bound the dose consequences for fuel cycles from 12 to 24 months.

3.1 Revised CR atmospheric dispersion factors and new core inventory

The licensee provided core inventories for the current analysis of record, the revised annual fuel cycle, and the 12 to 24 month fuel cycle. In accordance with the guidance in RG 1.183, the licensee used the ORIGEN-ARP computer code to calculate the revised core inventories. The NRC staff examined the core inventories, especially those radionuclides that dominate the dose analyses. As a result of this examination the NRC staff found that the use of the revised annual fuel cycle inventory proposed by the licensee provides the most conservative (bounding) dose consequences and, therefore, is acceptable for use in the revised LOCA analysis. Based on the above, the NRC staff finds that the guidance in RG 1.183 is met.

The current LOCA dose calculation methodology, which was based on an AST methodology in accordance with 10 CFR 50.67, "Accident source term," was approved by the NRC in September 2005 in Amendment 167 to Facility Operating License NPF-62. The Safety Evaluation related to license Amendment 167 (ADAMS Accession No. ML052570461) documents the staff's review and acceptance of the χ/Q values used in the current AST dose analyses.

The outside air is normally brought in through one of two outside air intakes that are located on the east and west sides of the plant. Pursuant to RG 1.194, if the ventilation system design allows the operator to manually select the least contaminated outside air intake as a source of outside air makeup and close the other intake, the χ/Q values for the favorable intake may be reduced by a factor of four to account for the dual inlet and the expectation that the operator will make the proper intake selection. This protocol should be used only if the dual intakes are in different wind direction windows and if there are redundant, engineered safety-feature-grade radiation monitors within each intake, with control room indication and alarm, to monitor the intakes.

However, subsequent to the issuance of Amendment 167 to the CPS operating license, EGC determined that the factor of four reduction in χ/Q values is no longer applicable because there are single failure issues with the outside air intake dampers. The licensee states in its letter dated January 9, 2018, that no credit is taken for the dual, separated outside air intake design feature for evaluating the radiological consequences of DBAs.

Table 1 in the Safety Evaluation related to license Amendment 167 (ADAMS Accession No. ML052570461) documents the staff's review and acceptance of the χ/Q values used in the current AST dose analysis. Those χ/Q values, along with the revised design basis χ/Q values proposed for use in this LAR, are listed in Table 1:

Table 1: Filtered Intake Control Room χ/Q Values (sec/m3)

Time Period	Current Design Basis			Proposed Revised Design Basis
	West Intake	East Intake	Most Favorable Intake Divided by Four	
0 - 2 hours	9.45E-04	9.75E-04	2.36E-04	9.45E-04
2 - 8 hours	7.58E-04	7.09E-04	1.77E-04	7.58E-04
8 - 24 hours	3.28E-04	2.93E-04	7.33E-05	3.28E-04
24 - 96 hours	2.61E-04	2.13E-04	5.33E-05	2.61E-04
96 - 720 hours	1.85E-04	1.79E-04	4.48E-05	1.85E-04

Table 1 shows that for the proposed revised design basis, the most favorable χ/Q value was selected for the 0-2 hour time period whereas the least favorable χ/Q values were selected for the remaining time periods. Request for Additional Information (RAI) RMET/RHM-1 (ADAMS Accession No. ML18242A156) was issued seeking justification as to why the χ/Q value for the limiting (least favorable) outside air intake was not used for 0-2 hour time interval for the post-LOCA control room dose calculations.

In its letter dated August 29, 2018 (ADAMS Accession No. ML18242A156) in response to a request for additional information, the licensee stated that a loss of divisional power would result in failure of either the east or west intake. For the revised LOCA analysis, the licensee used the χ/Q values from the intake which gave the higher overall control room dose from 0-720 hours. The licensee presented dose results using the east intake and west intake χ/Q values for the duration of the accident (0-720 hours) which showed χ/Q values for the west intake resulted in the bounding (higher) dose results.

Because a loss of divisional power would result in the loss of one or the other intake, CPS should be considered as having two separate/single remote intakes in this scenario; that is, either the east or west intake will be operable following a loss of divisional power. Because the west intake χ/Q values produce the higher control room dose for the duration of the accident, the NRC staff found the use of the west intake χ/Q values for all time intervals acceptable.

The NRC staff finds that the use of the west intake results in a conservative estimate of CR dose consequences and is, therefore, acceptable. The removal of the factor or four reduction factor and the use of the west intake corrects a misapplication of RG 1.194 and therefore the NRC staff concludes that the guidance in RG1.194 is met. In addition, the NRC staff finds that based on the above, the NRC staff's other conclusions in the NRC staff's SEs related to Amendments 167 and 210 to Facility Operating License No. NPF-62 for the CPS, Unit 1, are not affected and therefore the guidance in RG 1.23, RG 1.145, and RG 1.183 continue to be met.

In its letter dated January 9, 2018, Enclosure 5, the license provided the impact of the revised χ/Q and revised source term on the USAR, Chapter 15 Accident Analysis. The NRC staff finds that in all analyses, the CR dose, the Exclusion Area Boundary Dose, and the Low Population Zone dose were less than the 10 CFR 50.67 dose limits and the SRP dose guidance. Based on the above, the NRC staff concludes that the 10 CFR 50.67 requirements are met and the SRP, Section 2.3.4 guidance is met. Because the CR dose meets the 10 CFR 50.67 requirements and meets the SRP, Section 2.3.4 guidance, the NRC staff concludes that the 10 CFR 50, Appendix A, GDC 19, "Control Room," requirements are met.

3.2 Revised Feedwater Isolation Valve liquid and air leakage

The licensee revised the assumed value for the emergency core cooling system (ECCS) feedwater isolation valve leakage from the CLB value of 2 gpm to a value of 1.5 gpm for the 30 day duration of the accident analysis period. This change to the LOCA dose consequence analysis resulted in the necessity to revise the associated TS Surveillance Requirement (SR) 3.6.1.3.11 to verify that the combined leakage rate for both primary containment feedwater penetrations is less than or equal to 1.5 gpm when pressurized to larger than or equal to 1.1 P_a where P_a (pressure absolute) is the calculated peak containment internal pressure related to the design basis LOCA as specified in the TS.

Current SR 3.6.1.3.11:

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	--------------------------NOTE----------------------------- Only required to be met in MODES 1, 2, and 3. Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ 2 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.

Revised SR 3.6.1.3.11:

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	--------------------------NOTE----------------------------- Only required to be met in MODES 1, 2, and 3. Verify that the combined leakage rate for both primary containment feedwater penetrations is ≤ 1.5 gpm when pressurized to ≥ 1.1 P_a.	In accordance with the Primary Containment Leakage Rate Testing Program.

This change to a more restrictive leakage criterion represents an enhancement to the radiological safety of the plant and is, therefore, acceptable to the NRC staff.

The licensee is also proposing a reduction in feedwater isolation valve air leakage from 10.98 cfm to 8.64 cfm. Based on the USAR markup, (page 15.6-10, Item 4), provided in Enclosure 2 to its letter dated January 9, 2018, the licensee states that the containment atmosphere leakage (air) through the feedwater penetration applies in the first 1 hour and the liquid leakage through the feedwater penetration applies for the duration of the accident after 1 hour.

By Amendment No. 105, dated June 21, 1996 (ADAMS Accession No. ML020990619), CPS adopted Option B, "Performance Based Testing Requirements," of Appendix J to 10 CFR 50, "Primary Reactor Containment Leakage Rate Testing for Water Cooled Power Reactors." The exemptions to the requirements 10 CFR Part 50, Appendix J, are stated in Section 2.D of the CPS Facility Operating License. In addition, CPS TSs are contained in Appendix A to the Facility Operating License.

Appendix J to 10 CFR 50, Option B, Section III.B, "Performance-Based Requirements for Type B and C Tests," requires pneumatic tests to measure containment isolation valve leakage rates. NEI 94-01, Revision 3-A, Section 8.0, "Testing Methodologies for Type A, Type B and Type C Tests," requires Type A, Type B, and Type C tests should be performed using the technical methods and techniques specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS)-56.8–2002, "Containment System Leakage Testing Requirements" (available from http://www.ans.org/standards/), or other alternative testing methods that have been approved by the NRC. Section 2, "Definitions," of ANSI/ANS-56.8-2002 defines Type C test as "A pneumatic test to measure leakage rates from containment isolation valves, which are potential gaseous leakage pathways from containment during a design-basis LOCA."

The NRC staff requested additional information as follows:

> Describe the Appendix J leakage tests currently performed to satisfy the requirements of SR 3.6.1.3.11 and TS 5.5.13. If the tests, including the test medium, are different from the approved methods in Appendix J to 10 CFR 50 and ANSI/ANS-56.8-2002, please describe where the exemptions/exceptions are explicitly stated in the CPS license documents including any prior staff approval obtained for such exemptions/exceptions.
>
> Explain if the reduced liquid and containment atmosphere leakage values proposed by the LAR are already supported by recent historical results of Appendix J testing or if new testing would be necessary.

In its August 29, 2018, letter, the licensee responded to the NRC staff's request as follows:

> CPS implemented a feedwater leakage control system (FWLCS) in 2000 to provide an enhanced means of isolating the FW penetrations post-LOCA. With the operation of the FWLCS, the periodic leakage testing requirement for the primary containment FW penetration isolation valves is being conducted with a water leakage test in lieu of an air leakage test. The NRC has previously reviewed the CPS FWLCS modification and testing of valves with water as documented in a safety evaluation dated April 25, 2000, (i.e., Accession No. ML003710668).

> The CPS 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Program covers pneumatic tests and also contains provisions for water testing the FW isolation valves. The FW Containment Penetrations 1MC-09 and 1MC-10 are tested using a low pressure (i.e., not 1000 psig) water test in accordance with CPS Procedures 9861.05D013, "Local Leak Rate Test Data Sheet for 1MC009," and 9861.05D014, "Local Leak Rate Test Data Sheet for 1MC010." This testing is in accordance with ANS-56.8-2002, Section 3.4, "Qualified Seal System Testing Requirements." The qualified seal system is provided by the FWLCS as previously reviewed by the NRC as discussed above.

In addition, the licensee also provided recent historical results of Appendix J tests for the feedwater penetrations during the refueling outages in 2013, 2015, and 2017. The results indicate that the combined leakage for the feedwater penetrations varied from a low of 0.15 gpm to a high of 0.88 gpm versus the acceptance criterion of 2.0 gpm.

The exception to perform pneumatic testing in lieu of water testing was granted after the licensee adopted Option B, "Performance Based Testing or Water Cooled Reactors." Although never identified as an exception within the TSs, the NRC staff recognizes that the licensee has been testing the feedwater isolation valves with a water leakage test in lieu of an air leakage test since the implementation of FWLCE in 2000 as approved by the NRC. The test results during the recent outages for the combined leakage of the two feedwater penetrations show a considerable margin even when compared to the proposed reduced acceptance criterion of 1.5 gpm for the two penetrations. The NRC staff finds the licensee response acceptable.

Based on the above, the NRC staff finds that the proposed TS change to be acceptable. In addition, since the TS surveillance ensures that the licensee's accident source term is bounding the NRC staff finds that 10 CFR 50.36(c) is met because the proposed limit ensures that the CR HVAC maintains the necessary quality of the control room atmosphere.

3.3 Revised CR inleakage

The licensee reduced the assumed value for CR inleakage from the CLB value of 1,100 cfm to a value of 1,000 cfm. The NRC staff notes that licensee's typically assume a value for CR inleakage that is substantially higher than the actual measured values from CR inleakage testing. Periodic CR inleakage testing serves to assure that the actual inleakage will be less than the amount assumed in the dose consequence analysis. Therefore, reducing the value of the CR inleakage assumed in the dose consequence analysis (that still meets the requirements of 10 CFR 50.67) does not have an impact on the design characteristics of the CR and is therefore acceptable to the NRC staff. In Section 3.1, above, the NRC staff concluded that 10 CFR 50.67 is met.

3.4 Credit for mixing in 50 percent of the secondary containment volume

Enclosure 2 to the LAR contained a markup of the revised USAR and the TS bases pages. The markup in USAR, page 15.6-10, contained a new item (6) added to Section 15.6.5.5.1.2, "Fission Product Transport to the Environment." Item (6) states "Credit is taken for mixing primary containment leakage in 50 percent of the secondary containment volume in accordance with RG 1.183, Appendix A, Section 4.4."

Appendix A, Section 4.4, to RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," states in part:

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

The NRC staff requested that the licensee provide information regarding the means available for mixing of primary containment leakage with secondary containment volume and the supporting review/analysis performed at CPS to justify the 50 percent mixing assumption within secondary containment.

In its letter dated August 29, 2018, the licensee stated the following:

> The current Clinton Power Station (CPS) loss of coolant accident (LOCA) analysis credits mixing in 50 percent (%) of the Secondary Containment volume consistent with NRC Standard Review Plan (SRP) Section 6.5.3, "Fission Product Control Systems and Structures," and Regulatory Guide 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a [Light Water Reactor] Loss-Of-Coolant Accident," Section 4.4. The leakage from the Primary Containment to the Secondary Containment is 0.598%/day and the containment volume (excluding the Drywell) is 1,512,341 cubic feet (ft^3). The flow rate from the Primary Containment is therefore approximately six (6) cubic feet per minute (cfm) into a Secondary Containment volume of approximately 1,700,000 ft^3. The containment leakage, if any, would most likely be associated with piping penetrations, which are located in the lower part of the containment (e.g., Elevation (El.) 707'-6" mean sea level), or door seals. The small amount of containment leakage would have to diffuse through the secondary containment prior to being exhausted by the Standby Gas Treatment System (SGTS) to the environment. Significant mixing would occur as the leakage travels through the Secondary Containment prior to entering the SGTS. If power is available, the Secondary Containment heating, ventilation, and air conditioning (HVAC) system would provide additional mixing by supplying air from clean areas to potentially contaminated areas in accordance with good as-low-as-reasonably-achievable (ALARA) radiation protection practices. The Secondary Containment volume is 1,700,000 ft^3 (i.e., rounded down from 1,704,995 ft^3) so the mixing volume used in the analysis is 8.50E+05 ft^3.
>
> The CPS design includes the SGTS which is a safety-related HVAC system. Although this system is not a mixing system per se, the system does take suction directly or indirectly from every compartment in the Auxiliary Building which provides mixing. The following rooms located at El. 707'-6" are serviced by the SGTS (i.e., total flow of approximately 1000 cfm for these areas):
>
> • Low Pressure Core Spray (LPCS) Pump Room,
> • Residual Heat Removal (RHR) "A" Pump Room,
> • RHR "A" Heat Exchanger (Hx) Room,
> • Main Steam Tunnel,
> • RHR "B" Pump Room,
> • RHR "B" Hx Room,
> • RHR "C" Pump Room,
> • High Pressure Core Spray Pump Room, and
> • Accessible Area El. 715'-0"

The SGTS also services the following areas (i.e., total flow approximately 3000 cfm for these areas):

- Reactor Water Cleanup (RWCU) "A", "B", and "C" Pump Rooms, and
- Radwaste Pipe Tunnel
- Combustible Gas Control Boundary (various elevations)
- Auxiliary Building Aisles

Although no specific transport analysis was performed due to the unknown location of the potential leakage, there is sufficient justification for the assumption that the leakage is mixed in 50% of the Secondary Containment volume.

Based on the large secondary containment volume as compared to a very small amount of containment leakage, and the availability of SGTS suction directly or indirectly from every compartment in the secondary containment, the NRC staff finds that a reasonable basis exists for assuming significant mixing of the containment leakage in secondary containment. In addition, based on the licensee statement that "if power is available, the Secondary Containment HVAC system would provide additional mixing by supplying air from clean areas to potentially contaminated areas in accordance with good ALARA radiation protection practices," the NRC staff finds that there is additional consideration for significant mixing. As stated in USAR, Section 9.4.2, "Fuel Handling Building," the fuel handling HVAC serves certain areas of secondary containment during normal operating conditions. The fuel handling HVAC will be isolated from secondary containment by closing redundant isolation dampers on any signal initiating the SGTS. However, there are room recirculation cooling systems provided in certain areas within the secondary containment as indicated in USAR, Section 9.4.5.3, "ECCS Equipment Area Cooling," which operate during DBAs and provided with emergency power. Therefore, the recirculation cooling systems within secondary containment will have the ability to assist in mixing.

The licensee stated that no specific transport analysis was done due to unknown leakage locations. However, the licensee provided adequate qualitative/quantitative reasons such maximum leakage values being very small compared to secondary containment volume, SGTS suction locations distributed throughout secondary containment. Therefore, the NRC staff concludes that the 50 percent mixing assumption in the secondary containment is adequately supported at CPS.

Based on the above, the NRC staff concludes that the RG 1.183, Appendix A, Section 4.4, guidance is met.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the NRC staff notified the Illinois State official on October 29, 2018, of the proposed issuance of the amendments. The State official had no comments.

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR, Part 20, or change inspections or surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on March 13, 2018 (83 FR 10918), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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Date of issuance: January 3, 2019

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT XXX RGARDING REVISED ALTERNATIVE SOURCE TERM DOSE CALCULATION (EPID L-2018-LLA-0004) DATED JANUARY 3, 2019

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