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

September 6, 2010

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: APPLICATION OF ALTERNATIVE SOURCE TERM (TAC NOS. ME0068 AND ME0069)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 197 to Facility Operating License No. NPF-11 and Amendment No.184 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated October 23, 2008 (Agencywide Documents Access and Management System (ADAMS) Package No. ML083100153), as supplemented by letters dated September 28, 2009 (ADAMS Accession No. ML092710196), November 18, 2009 (ADAMS Accession No. ML093220838), March 29, 2010 (ADAMS Package No. ML100890060) and August 3, 2010 (ADAMS Package No. ML102230205).

The amendments revise the Technical Specifications to support the application of alternative source term methodology with respect to the loss-of-coolant accident and the fuel handling accident.

The September 28, 2009, November 18, 2009, March 29, 2010, and August 3, 2010, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

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

Sincerely,

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 197 to NPF-11
2. Amendment No. 184 to NPF-18
3. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. NPF-11

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated October 23, 2008, as supplemented by letters dated September 28, and November 18, 2009, March 29, and August 3, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: September 6, 2010

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. NPF-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated October 23, 2008, as supplemented by letters dated September 28, and November 18, 2009, March 29, and August 3, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: September 6, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 197 AND 184

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

DOCKET NOS. 50-373 AND 50-374

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

Remove	Insert
License NPF-11	License NPF-11
Page 3	Page 3
License NPF-18	License NPF-18
Page 3	Page 3
TSs	TSs
3.1.7-1	3.1.7-1
3.3.6.1-9	3.3.6.1-9
3.6.1.3-8	3.6.1.3-8
3.6.4.1-3	3.6.4.1-3
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15

Am. 146
01/12/01

(4) Exelon Generation Company, LLC, 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; and

Am. 146
01/12/01

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2.

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) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 194
08/28/09

(3) DELETED

Am. 194
08/28/09

(4) DELETED

Am. 194
08/28/09

(5) DELETED

Am. 194
08/28/09

(6) DELETED

Am. 194
08/28/09

(7) DELETED

Am. 34
12/08/87

(5) 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 LaSalle County Station, Units 1 and 2.

C. The 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:

Am. 125
05/09/00

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3489 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No 184 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 181
08/28/09

(3) DELETED

Am. 181
08/28/09

(4) DELETED

Am. 181
08/28/09

(5) DELETED

Am. 181
08/28/09

(6) DELETED

Am. 181
08/28/09

(7) DELETED

Am. 181
08/28/09

(8) DELETED

Am. 181
08/28/09

(9) DELETED

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours

(continued)

Table 3.3.6.1-1 (page 4 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RWCU System Isolation (continued)					
k. Reactor Vessel Water Level–Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ -58.0 inches
l. Standby Liquid Control System Initiation	1,2,3	2[b]	I	SR 3.3.6.1.5	NA
m. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA
5. RHR Shutdown Cooling System Isolation					
a. Reactor Vessel Water Level–Low, Level 3	3,4,5	2[c]	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 11.0 inches
b. Reactor Vessel Pressure–High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 143 psig
c. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA

(b) Only inputs into one of two trip systems.

(c) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through any one main steam line is ≤ 200 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	31 days
SR 3.6.4.1.3	Verify the secondary containment can be drawn down to ≥ 0.25 inch of vacuum water gauge in ≤ 900 seconds using one standby gas treatment (SGT) subsystem.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.4	Verify the secondary containment can be maintained ≥ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 4400 cfm.	24 months on a STAGGERED TEST BASIS for each SGT subsystem
SR 3.6.4.1.5	Verify all secondary containment equipment hatches are closed and sealed.	24 months

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed prior to startup following L2R12.

3. The potential valve atmospheric leakage paths that are not exposed to reverse direction test pressure shall be tested during the regularly scheduled Type A test. The program shall contain the list of the potential valve atmospheric leakage paths, leakage rate measurement method, and acceptance criteria. This exception shall be applicable only to valves that are not isolable from the primary containment free air space.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 39.9 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.0% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, the seal leakage rate is $\leq$ 5 scf per hour when the gap between the door seals is pressurized to $\geq$ 10 psig.

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals

5.5.14 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates; and

c. Actions to verify that the remaining cells are ≥ 2.07 V when a cell or cells have been found to be < 2.13 V.

5.5.15 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Area Filtration (CRAF) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body, or 5 rem TEDE, as applicable. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

(continued)

5.5 Programs and Manuals

5.5.15 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRAF System, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakge limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE NO. NPF-11

AND AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated October 23, 2008 (Agencywide Documents Access and Management System (ADAMS) Package No. ML083100153), as supplemented by letters dated September 28, 2009 (ADAMS Accession No. ML092710196), November 18, 2009 (ADAMS Accession No. ML093220838), March 29, 2010 (ADAMS Package No. ML100890060) and August 3, 2010 (ADAMS Package No. ML102230205), Exelon Generation Company, LLC (EGC, the licensee), requested changes to the technical specifications (TSs) for LaSalle County Station (LSCS), Units 1 and 2. The proposed changes would revise the TSs to support the application of alternative source term (AST) methodology with respect to the loss-of-coolant accident (LOCA) and the fuel-handling accident.

The September 28, and November 18, 2009, March 29, and August 3, 2010, supplements, contained clarifying information, did not expand the scope of the license amendment, and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

2.0 EVALUATION

2.1 Mechanical and Civil Engineering

2.1.1 Regulatory Evaluation

The following requirements and guidance documents are applicable to the NRC staff's review for Section 2.1 of this safety evaluation:

Pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 67 (10 CFR 50.67), "Accident source term," a licensee may revise its current accident source term by re-evaluating the consequences of design basis accidents (DBAs) with the AST. Additionally, Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," requires that structures, systems, and components (SSCs) necessary to assure the capability of the plant to mitigate the consequences of accidents, which could result in exposures comparable to the

guideline exposures provided in 10 CFR Part 100, be designed to remain in or remain functional during and after a safe shutdown earthquake (SSE). The NRC staff's review in the area of mechanical and civil engineering mainly focuses on the structural integrity, including seismic qualification, of SSCs such as the main steam isolation valve (MSIV) leakage bypass piping and other piping systems, electrical equipment, and heating, ventilation, and air conditioning (HVAC) system components (including the Control Room (CR) and Auxiliary Electric Equipment Room (AEER) HVAC system components) which are credited in the implementation of the AST at LSCS.

The NRC staff's evaluation considered General Design Criteria (GDC) 1, "Quality Standards and Records," and GDC 2, "Design Bases for Protection Against Natural Phenomena," which are provided in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of affected SSCs under postulated accident conditions, as analyzed with the implementation of an AST at LSCS. The acceptance criteria are based on the continued conformance with the requirements of the following regulations: 10 CFR Part 50, 50.55a, "Codes and standards," and GDC 1, as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed and GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of earthquakes combined with the effects of accident conditions.

The guidance associated with the implementation of an AST is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." With respect to the mechanical and civil engineering aspects of the AST implementation, RG 1.183 states that licensees must evaluate non-radiological impacts on a facility which are a consequence of the implementation of an AST methodology. For this particular AST amendment request, the licensee is requesting to implement a full scope AST at LSCS, as described in RG 1.183. A full scope AST implementation refers to the licensee's request to recalculate the dose consequences of select DBAs to address all five characteristics of the AST (i.e. the composition, magnitude, chemical and physical forms of the radioactive material and the timing of the material's release). For a full scope AST implementation, RG 1.183 states that the DBA LOCA must be re-analyzed at a minimum.

Additional guidance for the review can also be found in Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term," of the Standard Review Plan (SRP). Guidance regarding topics specific to the NRC staff's review of AST amendment requests can also be found in the General Electric (GE) Boiling-Water Reactor Owners Group (BWROG) Topical Report (TR) NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems" (Reference 3, Volume II). The NRC staff's SE for the BWROG TR, contained within Volume I of Reference 4, documents the staff's approval of this TR and provides additional guidance for licensees regarding the topic of MSIV leakage pathways.

2.1.2 Technical Evaluation

2.1.2.1 Main Steam Isolation Valve Alternate Leakage Treatment (ALT) Pathway

In performing the re-evaluation of the design-basis LOCA to support implementation of the AST at LSCS, the licensee indicated that credit would be taken for the reduction of the amount of radioactivity released through the MSIV leakage by deposition and holdup of fission products in the main steam line piping system and the main condenser at LSCS, as released during the LOCA DBA. In accordance with Section 4.5 of Appendix A in RG 1.183, crediting of deposition for leakage which bypasses secondary containment is allowed on a case-by-case basis. As indicated in Table 3.6-1 in Attachment 1 of Reference 1, the MSIV ALT pathway is the only leakage pathway credited in the proposed AST that could bypass secondary containment.

The licensee is proposing to increase the amount of allowable leakage through one MSIV from 100 scfh (standard cubic feet per hour) to 200 scfh, while maintaining the total limit of 400 scfh for all four steam lines. The single line leakage limit was endorsed by the NRC in Volume I of Reference 3, which approved the use of BWROG TR NEDC-31858P (Volume II of Reference 3) for increasing MSIV leakage rates and deleting MSIV leakage control systems (LCS) in BWR plants which elected to employ the methodologies described within BWROG TR NEDC-31858P. With regards to the accreditation of components used for the deposition and holdup of MSIV leakage, Position 6.5 of Appendix A (LOCAs) of RG 1.183 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 deletion of the MSIV LCS in BWR plants was a generic movement based on issues identified with MSIV leakage rates. The BWROG TR provides a generic approach for demonstrating the seismic ruggedness of proposed ALT pathways, which had not previously been seismically analyzed for BWR plants in support of the deletion of these LCS.

The licensee cited License Amendments 112 and 97 (Reference 4) for LSCS Units 1 and 2 respectively, regarding the seismic ruggedness of the ALT pathway (also referred to as the MSIV-Isolated Condenser Leakage Treatment Method (MSIV-ICLTM)), which was credited in the proposed AST. Approved on April 5, 1996, these two amendments revised the LSCS Unit 1 and Unit 2 TSs to reflect the deletion of the LCSs at LSCS, Units 1 and 2, which were previously utilized to contain leakage that passed through the MSIVs on each unit's four main steam lines. Physically, the Unit 1 LCS was abandoned in place while the Unit 2 LCS was removed entirely. The deletion of the LCS was in concert with the establishment of the ALT pathway mentioned above, which consists of the four main steam lines, main steam drain lines, and the main condenser in each unit. As such, the ALT pathway became the primary pathway for leakage past the MSIVs. More significantly, the ALT pathway became the primary holdup point for fission products released as a consequence of a design-basis LOCA. Thus, LSCS was required to demonstrate the seismic ruggedness of the ALT pathway, including the main condenser, main steam lines, and the main steam drain lines, in support of the license amendment requests which were subsequently approved in Reference 4. It is noted in Section 3.6.7.5 of

Attachment 1 to Reference 1 that the main steam drain lines are not credited for deposition and hold up of any MSIV leakage for the purposes of the LOCA re-analysis in the proposed AST.

In support of the deletion of their LCS, LSCS did not utilize the generic methodologies presented in Reference 3, which focused on the use of an earthquake experience database, to seismically qualify ALT pathway components. Instead, the licensee opted to perform a plant-specific method of qualification, utilizing a complete analytic evaluation of the seismic adequacy of the ALT pathway piping, connected components, the main condenser, and supports. Seismic walkdowns were performed in support of the seismic ruggedness demonstration to identify conditions of piping and support configurations which would have resulted in seismically-induced pressure boundary failure and fission product release from the main steam line piping and main stream line drain piping. Additionally, the licensee analytically demonstrated the seismic adequacy of the turbine building, based on the fact that it contained many of the piping runs and pipe supports included in the ALT pathway. As previously stated, the NRC staff concluded that the methodology utilized by the licensee, including the analytical evaluations and seismic walkdowns, for determining the seismic ruggedness of the ALT pathway at LSCS was found acceptable (Reference 4). Additionally, the licensee confirmed, in response to the NRC staff's request for additional information (RAI) in Reference 2, that no modifications to the ALT pathways would be necessary to support the implementation of the proposed AST implementation. Thus, no changes are needed to the conclusions surrounding the seismic adequacy of the ALT pathway.

Based on the conclusion by the NRC staff in Reference 4 that the ALT pathways utilized at LSCS were demonstrated to be seismically adequate and no modifications to the ALT pathways are necessary to support the proposed AST implementation, the NRC staff concludes that the licensee has provided reasonable assurance that the ALT pathways are capable of performing their safety function during and following an SSE.

2.1.2.2 Standby Liquid Control (SLC) System Seismic Evaluation

In performing the re-evaluation of the LOCA DBA, the licensee indicated in Attachment 1 to Reference 1 that credit would be taken for controlling the pH in the suppression pool following a LOCA by injecting sodium pentaborate into the reactor core, utilizing the SLC system. Detailed design information regarding the SLC system at LSCS can be found in Section 9.3.5, "Standby Liquid Control System (BWRs)," of the facility's Updated Final Safety Analysis Report (UFSAR). To demonstrate that the SLC system is capable of performing its intended safety function during a LOCA following AST implementation, the licensee addressed the guidance provided by the NRC staff in Reference 5. This guidance provides four review guidelines which the licensee should use to evaluate whether their plant-specific SLC system can be credited as either safety-related or comparable to a safety-related system for the purposes of controlling the pH of the reactor coolant following a LOCA. Review guidelines 1 and 4 are addressed below, while the remaining guidelines are addressed in Section 2.3.2 of this safety evaluation report.

With respect to the first guideline found in Reference 5, the licensee indicated that the SLC system at LSCS was classified as a safety-related system. Given this classification, the SLC system at LSCS is designed to Seismic Class I requirements based on the seismic qualification methodologies found in Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the LSCS' UFSAR. Additionally, the licensee referenced the redundancy of the active components within the SLC system, which is addressed by the fourth guideline of

Reference 5. This guideline requests licensees to address any non-redundant, active components in detail, using one of three response options. It was stated on Page 35 and 36 of Attachment 1 to Reference 1, that the only active non-redundant components within the SLC system are the SLC system discharge header to reactor pressure vessel (RPV) inboard and outboard check valves (1(2)C41-F007 and 1(2)C41-F006, respectively) in each unit (parentheses designate appropriate unit). In addressing the non-redundant nature of these valves, the licensee provided the requested information regarding the design-basis conditions which these valves may be required to operate under, including environmental and seismic conditions. As previously indicated, the entire SLC system for LSCS was designed to meet Class I seismic qualification requirements, including the aforementioned check valves. The valves were indicated to be essential components, which were designed to Class I seismic requirements, thus satisfying the information requested by the guidance above.

Based on the information provided by the licensee and the design basis information for the LSCS' SLC system, denoting that the system was designed and qualified to Seismic Class I requirements, the NRC staff finds the licensee's assessment of the aforementioned system acceptable with respect to its ability to perform its credited safety function upon implementation of the proposed AST.

2.1.2.3 HVAC Systems Evaluation

In Section 3.6 of Attachment 1 to Reference 1, the licensee indicated that credit would be taken for portions of the Control Room HVAC and AEER systems for the purposes of reanalyzing the dose consequences at the LSCS in support of the proposed AST. The control room envelope (CRE) at LSCS is made up of both the CR and AEER which share the filtered emergency makeup system, but have separate filtered recirculation systems. In response to the NRC staff's RAI, which sought to clarify whether any modifications were being made to these HVAC systems and sought to clarify the accreditation of these systems for the proposed AST, with respect to the seismic qualification of these systems, the licensee provided additional details pertaining to these topics (Reference 2). The licensee confirmed that no modifications would be made to the existing equipment for the CR and AEER HVAC systems. Additionally, the licensee confirmed that the portions of the CR and AEER HVAC systems, which are being credited in the AST analyses, were designed and installed to meet the requirements for Seismic Category I components. As such, these design requirements provide reasonable assurance that the aforementioned system components will continue to perform their safety function during and after a design basis seismic event, i.e. a SSE.

The licensee did indicate that the heating equipment is the only portion of the CR and AEER HVAC systems was not designed and installed to Seismic Category I requirements. This equipment is not credited as performing any safety-related functions with respect to the dose reducing, safety-related functions performed by the portions of the CR and AEER HVAC systems, which are credited for the proposed AST. However, in the same response to the NRC staff's RAI noted above, the licensee did indicate that the heating equipment is seismically supported (as stated in Section 9.4.1, "Control Room Area Ventilation Systems," of the LSCS UFSAR), and, thus will not adversely impact the function of the safety-related components of these systems during and after a design basis seismic event. The design bases for these systems can be found in Section 9.4.1 of the LSCS UFSAR. The specific information surrounding the seismic design methodologies which were used to qualify these systems can be

found in Sections 3.7, "Seismic Design," through 3.10, "Seismic Qualification of Seismic Category I Instrumentation and electrical Equipment," of the LSCS' UFSAR.

The NRC staff finds the licensee's assessment that these systems will continue to operate safely upon implementation of the proposed AST, acceptable based on the above discussion, which indicates that there is reasonable assurance that the credited components of the CR and AEER HVAC systems would be structurally capable of performing their intended functions under a design basis seismic event.

2.1.2.4 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed license amendment request (LAR) associated with the implementation of the full scope AST methodology at LSCS on portions of the ALT Pathway, the SLC system, and HVAC system components with regards to the seismic qualification involved, with these items as they relate to the AST implementation. On the basis of the NRC staff's review as described above, that demonstrates the seismic adequacy of all the aforementioned components, the NRC staff finds that the proposed AST implementation will not have an adverse impact on the structural ability of these systems to withstand and perform their intended functions when subjected to a design-basis seismic event. Therefore, with respect to the structural integrity of the components described here within, the NRC staff finds the proposed amendment to revise the TS's associated with the implementation of a full scope AST methodology, as it applies to the LOCA and fuel-handling accident (FHA) DBA analyses at LSCS, acceptable.

2.2 Materials and Chemical Engineering

2.2.1 Regulatory Evaluation

The following requirements and guidance documents are applicable to the NRC staff's review for Section 2.2 of this safety evaluation:

Implementation of the AST by the licensee required re-analyzing several design-basis accidents using new source terms. The licensee performed these tasks by following the requirements of 10 CFR 50.67, "Accident source term." It also applied for a license amendment under 10 CFR 50.90, "Application for amendment of license or construction permit." An acceptable accident source term is a permissible amount of radioactive material that could be released to the containment from the damaged core following an accident. Because of improved understanding of the mechanisms of the release of radioactivity, the current AST could be replaced by a less restrictive AST. However, this replacement is subject to performing a successful re-evaluation of the major design-basis accidents. The guidance for implementation of an AST is provided in RG 1.183.

The NRC staff reviewed the portion of the amendment dealing with the licensee's analysis of maintaining suppression pool pH ≥7 for 30 days following a LOCA. According to RG 1.183, maintaining a basic pH will minimize re-evolution of iodine from the suppression pool water.

2.2.2 Technical Evaluation

After a LOCA, a variety of different chemical species are released from the damaged core, including radioactive iodine. This iodine, if released to the outside environment, can significantly contribute to radiation doses; therefore, it is essential to keep as much of it as possible confined within the plant's containment. According to RG 1.183, of all the radioactive iodine released from the reactor coolant system into containment, 95 percent is assumed to be released in ionic form as cesium iodide (CsI), 4.85 percent is assumed to be released as elemental iodine (I_2) and 0.15 percent is assumed to be released as hydriodic acid (HI). CsI and HI are ionized in water environments and, therefore, soluble; however, elemental iodine is insoluble. Because elemental iodine in insoluable, it is important to maintain the released iodine in its ionic form in order to reduce any exposure. Unfortunately, in the post-accident radiation environments existing in containment, radiolysis will convert some of the ionic iodine dissolved in water into the elemental form. Fortunately, the degree of conversion to the elemental form decreases with increasing pH and at a pH ≥ 7, it becomes very small (<1 percent). The relationship between the degree of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control."

In LSCS Units 1 and 2, most of the iodine is released from the core to the suppression pool. Therefore, in order to keep it dissolved in its ionic form, the suppression pool water should be kept at pH ≥ 7 throughout the 30-day post-LOCA period. The licensee has calculated that, because of strong acid formation in the containment, this is not achievable without adding buffering chemicals to control the suppression pool's pH.

The licensee calculated that after a LOCA, the pH value will be continuously decreasing due to formation of hydrochloric and nitric acid in containment. Hydrochloric acid is formed from the decomposition of Hypalon cable insulation by radiation. About 1.61×10^{-3} g-mols/liter of hydrochloric acid is formed during the 30-day period. Nitric acid is formed by irradiation of air and water and about 6.74×10^{-5} g-mols/liter of nitric acid is formed during the same 30-day period. Both are strong acids and will significantly contribute to lowering suppression pool pH. In order to neutralize the acids' effect, the licensee used the buffering effect of sodium pentaborate from the SLC system. The main purpose of the SLC is to control reactivity in the case of control rod failure. However, since sodium pentaborate is derived from a strong base and a weak acid it also acts as a buffer. Such buffering action can maintain basic pH in the suppression pool despite the presence of strong acids. The licensee calculated that a 1001.1 lbs minimum addition of sodium pentaborate from the SLC system would maintain a pH > 7 in the suppression pool for 30 days (ADAMS Accession No. ML083100204). In the licensee's analysis, the SLC system boron addition is accomplished 3.4 hours after the start of the LOCA, which was just under the maximum allowable time of 3.5 hours that maintained the suppression pool pH > 7. The 3.4 hour delay included a 2-hour delay in the initiation of the ECCS and the SLC system; having only the minimum emergency diesel power available, and the minimum SLC system injection flow rate.

In order to evaluate the beneficial effect of sodium pentaborate, the licensee calculated suppression pool pH for unbuffered and buffered cases. As expected, without addition of sodium pentaborate, the value of pH during the 30-day period was below 7, reaching a minimum pH value of 2.8. The addition of the sodium pentaborate increases the suppression pool pH above 7.4 for the 30 days post-LOCA.

The NRC staff reviewed the licensee's analysis and found it acceptable. The NRC staff also independently verified the licensee's calculations for acid formation and the amount of buffer needed to maintain the suppression pool pH > 7 for 30 days post-LOCA. The NRC staff also assumed a time of 3.4 hours to complete the sodium pentaborate addition. The NRC staff finds that by using sodium pentaborate as a buffer, the pH of the suppression pool will remain above a pH of 7 for 30 days post-LOCA.

2.2.3 Conclusion

In the submittal, the licensee described its methodology for maintaining the suppression pool's pH above 7 for the 30 days following a LOCA. The methodology relies on using the buffering action of sodium pentaborate, introduced into the suppression pool from the SLC system. The licensee provided analyses justifying that the minimum calculated injection of 1001.1 lbs of sodium pentaborate will ensure that the pH in the suppression pool will stay above 7 for 30 days after a LOCA. The NRC staff has reviewed the calculations and justifications provided by the licensee. The staff finds the analysis acceptable as presented in the licensee's submittal, which indicates that the suppression pool pH will stay basic for the period of 30 days after a LOCA.

2.3 Containment and Ventilation

2.3.1 Regulatory Evaluation

This safety evaluation input addresses the impact of the proposed changes on previously analyzed design basis accident radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, 10 CFR Part 50 Appendix A, GDC-19, "Control Room", and Standard Review Plans (SRP) (NUREG-0800). Except where the licensee has proposed a suitable alternative, the following requirements and guidance documents are applicable to the NRC staff's review for Section 2.3 of this safety evaluation:

- RG 1.183

- SRP Section 6.4, "Control Room Habitability Systems" (with regard to control room meteorology)

- SRP Section 6.5.1, "ESF Atmosphere Cleanup Systems"

- SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term"

- NRC Review Guidelines, "Guidance on the Assessment of a BWR LC System for pH Control," dated February 12, 2004 (ADAMS Accession No. ML040640364)

2.3.2 TS Section 3.1.7, "Standby Liquid Control (SLC) System"

The proposed change revises the applicability of TS Section 3.1.7 to add the requirement for the limiting condition for operation (LCO) to be met in Mode 3. This change implements AST assumptions regarding the use of the SLC system to buffer the suppression pool following a LOCA involving significant fission product release. The proposed change revises Condition C Required Actions to add an additional requirement, C.2, to be in Mode 4 with a completion time of 36 hours.

LSCS evaluated the suppression pool pH over the 30-day duration of the DBA LOCA and demonstrated that with injection of sodium pentaborate through the SLC system, the pH will remain above 7.0. Therefore, iodine conversion to elemental with re-evolution is considered inconsequential in the LOCA calculation. The control of pH also significantly limits the potential for airborne release from sub-cooled ECCS leakage inside and outside of secondary containment.

LSCS proposes to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. The SLC system design was not previously reviewed for this safety function (i.e., pH control post-LOCA).

The SLC system consists of the boron solution tank, the test water tank, two positive displacement pumps, two explosive valves, two motor-operated pump suction valves, and associated local valves and controls are located in the secondary containment. The liquid is piped into the reactor vessel and discharged near the bottom of the core shroud.

LSCS states that the SLC system is a safety-related system and meets the following requirements:

- The SLC system is provided with standby alternating current (ac) power supplemented by the emergency diesel generators.

- The SLC system is seismically qualified in accordance with RG 1.29, "Seismic Design Classification," and Appendix A to 10 CFR Part 100.

- The SLC system is incorporated into the plant's American Society of Mechanical Engineers Code inservice inspection and inservice testing programs based upon the plant's code of record (10 CFR 50.55a).

- The SLC system is incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."

- The SLC system meets 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear power plants," and Appendix A (GDC 4, "Environmental and Dynamic Effects Design Bases") to 10 CFR Part 50.

The LSCS SLC system activation steps are in a safety-related plant procedure (i.e., Emergency Operating Procedure (EOP) LGA-001, "RPV Control"). LSCS states that they will revise

LGA-001, which will ensure that SLC system injection is started from the boron solution storage tank during a DBA LOCA. In addition, LGA-001 will be revised to ensure no steps would terminate the injection during a DBA LOCA prior to emptying the SLC storage tank (i.e., injection of the full content into the RPV). This ensures complete injection upon a LOCA signal.

The license amendment request from EGC did not discuss how they would assure the sodium pentaborate injected below the core plate and core shroud will exit the reactor vessel through a postulated break above the core plate/shroud. In a letter dated August 27, 2009 (ADAMS Accession No. ML092330691), the NRC staff asked for additional information regarding the flow path from the SLC injection sparger through the reactor and out the postulated break and then to the suppression pool.

EGC responded in a letter dated September 28, 2009, that core spray and the low-pressure core injection (LPCI) systems will be in use. In an attachment, they indicated the flow path for the LPCI/core spray water is down through the core, mixing with the sodium pentaborate, flowing up through the jet pumps into the annular space between the reactor and the core shroud and out the recirculation suction pipe to the postulated break. Calculation L-003064, Revision 1, stated the flow would rise through the core following post-LOCA injection.

EGC provided additional information in a letter dated March 29, 2010. The letter clarified that all ECCS injection flow is inside the core shroud above the fuel. EGC also submitted Calculation L-003064, Revision 2. The revised calculation corrected a statement from Section 4.5 in Revision 1 that stated the sodium pentaborate rises through the core following post-LOCA injection.

Appendix D of RG 1.83 addresses assumptions for evaluating the radiological consequences of a BWR main steamline break accident. The NRC staff looked at the ability for sodium pentaborate injected through the SLC system to reach the suppression pool through a main steamline break.

EGC stated that the license basis for LSCS does not assume core damage with a main steamline break in containment upstream of the flow restrictors. LSCS UFSAR Chapter 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," indicates UFSAR Sections 6.2, "Containment Systems," and 6.3, "Emergency Core Cooling Systems," address main steamline breaks upstream of the flow restrictors. These sections document that the core will remain in a coolable geometry and that core damage will not occur during a main steamline break upstream of the flow restrictors. Based on no core damage sodium pentaborate injected through the SLC system is not required to exit the reactor vessel through the steamline break.

The steps that require activation of the SLC system are based upon symptoms of imminent or actual core damage. LSCS states when RPV water level drops below -150 inches, as read on the wide range level instruments, operator action will be to initiate SLC system injection from the SLC solution tank. This is indicative of a LOCA and that core uncovery is imminent and is symptomatic of core damage potential.

LSCS states that the instruments used to provide this indication are the wide range level instruments, which are listed in LSCS TS 3.3.3.1, "Post Accident Monitoring Instrumentation." These instruments are classified as Type A variable components as defined by RG 1.97,

"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Table 1, "Design and Qualification Criteria for Instrumentation." The post accident monitoring instrumentation LCO ensures the operability of RG 1.97, Type A, variables so that the control room staff can: (1) perform the diagnosis specified in the EOPs; and (2) take the specified, preplanned, manually-controlled actions for which no automatic control is provided, which are required for safety systems to accomplish their safety function.

LSCS states that Licensed Operators receive initial and periodic refresher training on the SLC system, and consequently, the steps that direct initiation of SLC.

A sufficient concentration and quantity of sodium pentaborate should be available for injection into the reactor vessel to control pH in the suppression pool. LSCS provided a copy of calculation L-003064, "Suppression Pool pH Calculation for Alternative Source Terms" as Attachment 8 to the LAR. This calculation provides the assumptions, inputs, methods, and results that demonstrate a sufficient concentration and quantity of sodium pentaborate will be available for injection into the reactor vessel to control pH in the suppression pool. Section 4.5 of the calculation discusses the adequacy of recirculation between the suppression pool and the reactor vessel through flowout break to provide transport and mixing.

According to UFSAR Section 9.3.5.2, "System Description," either of two key-locked switches (system A or B) on the control room console actuates the SLC system. This assures that switching from the "STOP" position is a deliberate act. Switching from "STOP" actuates the appropriate system, starts the appropriate injection pump, actuates both of the explosive valves, opens both pump suction motor-operated valves, and closes the reactor cleanup system outboard isolation valve to prevent loss or dilution of the boron.

If the pump lights, or explosive valve light indicates that the liquid may not be flowing, the operator can immediately utilize the alternate pump by turning its respective key-locked switch. Cross piping and check valves assure a flow path through either pump and either explosive valve. The local switch does not have a "STOP" position. This prevents the isolation of the pump from the control room. Pump discharge pressure is indicated in the control room. Based on the above, there is redundancy for starting the SLC system.

The LSCS SLC system cannot be considered redundant with respect to its active components. SLC System Discharge Header to RPV Outboard Check Valves - 1(2)C41-F006 and SLC System Discharge Header to RPV Inboard Check Valves - 1(2)C41-F007 are non-redundant active components that provide a flow path.

Therefore, LSCS proposes to demonstrate that this lack of redundancy is offset by satisfying Review Guideline 4(a) of Reference 5. Consistent with Review Guideline 2(a) of Reference 5, the following information is provided to describe the LSCS procedures for injecting sodium pentaborate using the SLC system.

SLC System Discharge Header to RPV Outboard Check Valves - 1(2)C41 -F006 and the SLC System Discharge Header to RPV Inboard Check Valves - 1(2)C41-F007.

(1) Manufacturer: Rockwell/Edward Model Number: 1-1/2-3674F316T(1)

(2) Worst-case accident conditions = 145 °F (1(2)C41 -F006)
Maximum accident temperature = 340 °F (1(2)C41 -F007)
Maximum accident pressure = 15 psia (1(2)C41 -F006)
Maximum accident pressure = 60 psia (1(2)C41 -F007)
Relative humidity = 100%
100 day LOCA dose = 1.0 x 107 rads (1(2)C41 -F006)
100 day LOCA dose = 2.0 x 108 rads (1(2)C41 -F007)
Seismic condition = maximum credible earthquake

(3) The components were purchased in accordance with 10 CFR Part 50, Appendix B.

(4) There was one LSCS Unit 1 (1C41 -F006) local leak rate test (LLRT) failure that required seat refurbishment due to leakage. No failures have occurred at LSCS in the forward direction of the check valves. No valve failures for other reasons have occurred for these Unit 1 and 2 check valves. A search of industry databases identified LLRT failures, similar to the LSCS Unit 1 LLRT failure, for the containment check valves. This type of failure does not impact the injection capability of the SLC system. No issues associated with the valves failing to open were identified.

There have been no LLRT failures at LSCS for inboard containment isolation purposes for these check valves (1(2)C41-F007). No valve failures for other reasons have occurred for these Unit 1 or 2 check valves.

In a letter dated August 27, 2009, the NRC staff asked for additional information regarding the scope of the foreign material control program for these check valves. EGC responded in a letter dated September 28, 2009, that the SLC system check valves will be designated as foreign material exclusion area (FEMA) 1. FEMA 1 is the highest level of foreign material exclusion (FME) imposed on a component or system under the LSCS FME program.

In summary, EGC has determined that the 1(2)C41-F006 and 1(2)C41 -F007 check valves have an acceptable performance history at LSCS.

(5) LSCS SR 3.1.7.8 requires verification of flow through one SLC subsystem from the pump into the RPV. The Frequency of surveillance requirement (SR) 3.1.7.8 is 24 months on a staggered test basis. EGC's procedure that implements this SR requires confirmation of flow that is > 41.2 gpm in the forward direction of the check valve.

(6) In the unlikely event that a SLC system injection path check valve fails to open, there are means of injecting sodium pentaborate using the reactor water cleanup (RWCU) system. Sodium pentaborate injection via the RWCU system is currently used for other events, such as anticipated transient without scram. Although the RWCU system could potentially be available for use, the AST analysis for LSCS does not credit this alternative method for pH control. Given the reliability of the non-redundant check valves of the SLC system, EGC concluded that compensating actions are not warranted.

The NRC staff reviewed the proposed changes for the SLC system. NRC Review Guidelines, "Guidance on the Assessment of a BWR SLC System for pH Control," dated February 12, 2004

(Reference 5) was used. In addition, RG 1.183, "Alternative Radiological Source Terms For Evaluating Design-Basis Accidents At Nuclear Power Reactors," NUREG-0800 SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms", and LSCS UFSAR Sections 6.2, 6.3, 9.3.5, and 15.6.5 were used in the review.

In a letter dated August 27, 2009, the NRC staff asked for additional information regarding how adequate mass of sodium pentaborate reaching the suppression pool is assured. In a letter dated March 29, 2010, EGC indicated the ECCS injection ensures a volumetric change rate of the affected regions of the reactor is approximately 18 changes per hour. The volumetric change rate in the suppression pool is roughly 0.7 changes per hour.

Based on the above, the NRC staff's assessment is that there will be sufficient flow of sodium pentaborate into the reactor vessel and from the vessel to the suppression pool, and therefore, the proposed change is acceptable. In addition, based on the review provided above and the conclusions outlined in Section 2.1.2.2 of this SER, the NRC staff considers the licensee's assessment of the effects of the AST implementation on the LSCS SLC system to be acceptable based on the intent of the four guidelines provided in Reference 5 having been met.

2.3.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

LSCS proposes to revise Table 3.3.6.1-1 of TS Section 3.3.6.1. This table lists, in part, the applicability requirements for primary containment isolation instrumentation. The proposed change revises the applicability of the SLC system initiation function of the RWCU system isolation instrumentation to add the requirement for this function to be operable in Mode 3. The revised applicability for this function is consistent with the revised applicability for the SLC system.

The addition of the applicability for Mode 3 (hot shutdown) will help assure that the suppression pool is maintained at a pH of 7 or higher in the event of a LOCA while in Mode 3 and is therefore acceptable.

2.3.4 TS Section 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

The proposed change revises SR 3.6.1.3.10 to increase the leakage limit through any one main steamline. Currently, the SR requires verification that the leakage rate through any one main steamline is less than or equal to 100 scfh, and that the leakage rate through all four main steam lines is less than or equal to 400 scfh, when tested at greater than or equal to 25.0 psig. The proposed change increases the leakage limit through any one main steam line from 100 scfh to 200 scfh. The combined leakage rate limit through all four main steamlines is not being changed. The revised SR 3.6.1.3.10 reads: "Verify leakage rate through any one main steamline is < 200 scfh and through all four main steamlines is < 400 scfh when tested at > 25.0 psig."

The Frequency for SR 3.6.1.3.10 is "In accordance with the Primary Containment Leakage Rate Testing Program," and this Frequency is not being changed.

The NRC staff's reviewed the licensee's proposal to increase the leakage limit through one main steamline to 200 scfh while maintaining the overall combined limit to 400 scfh and concluded that individual main steamline leakage limits are not needed to meet plant safety analyses. In

addition, the disadvantages associated with maintaining relatively low individual MSIV leakage rates are not justified by any additional conservatism the individual limits may provide. It is noted that the reworking of a single MSIV to meet the low individual leakage limits increases the probability of maintenance-induced defects and results in increased occupational radiation exposure. Based on the above, the NRC staff finds the proposed changes to TS 3.6.1.3 to be acceptable.

2.3.5 TS Section 3.6.4.1, "Secondary Containment"

The licensee proposes to revise SR 3.6.4.1.3 to increase the secondary containment drawdown time from less than or equal to 300 seconds to less than or equal to 900 seconds. This change reflects the application of AST assumptions.

The staff's assessment found the change to the secondary containment drawdown time acceptable because the value was increased for conservatism and supports the assumptions made in the licensee's revised DBA analyses under the proposed AST.

2.3.6 TS Section 5.5.13, "Primary Containment Leakage Rate Testing Program"

The licensee proposes to increases the maximum allowable primary containment leakage rate, L_a, at P_a, from 0.635 percent to 1.0 percent of primary containment air weight per day. The testing program, paragraph c, will be changed to read, "The maximum allowable primary containment leakage rate, L_a, at P_a is 1.0 percent of containment air weight per day."

The NRC staff's assessment found the change in maximum allowable primary containment leakage rate acceptable because the value was increased for conservatism and supports the assumptions made in the licensee's revised DBA analyses under the proposed AST.

2.3.7 TS Section 5.5.15, "Control Room Envelope Habitability Program"

The proposed change revises TS Section 5.5.15 to reflect that, with the adoption of AST methodology, the CR dose acceptance criterion for the LOCA and FHA are expressed in terms of total effective dose equivalent (TEDE).

The specific change revises TS 5.5.15, first paragraph, second sentence, to read; "The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body, *or 5 rem TEDE, as applicable*." (Proposed change is shown in italics). The addition of the reference to TEDE is consistent with the terminology used in 10 CFR 50.67. The NRC staff finds the revision acceptable.

2.3.8 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed license amendment request. The staff finds the proposed changes evaluated in Section 2.3 of this safety evaluation satisfy the relevant regulatory requirements (e.g. 10 CFR 50.67, GDCs) consistent with the guidance in RG 1.183, SRP 6.4, and SRP 6.5.1.

2.4 Health Physics and Human Performance

2.4.1 Regulatory Evaluation

The following requirements are applicable to the NRC staff's review for Section 2.4 of this safety evaluation:

Post-TMI Action Plan Requirements (NUREG-0737) were transmitted to all operating reactors licensees and construction permit holders by letter from Darrell G. Eisenhut dated October 31, 1980. Item II.B.2 of this Action Plan requires in part that licensees perform a radiation and shielding design review to ensure that personnel can access vital areas of the plant under accident conditions. The plant design is acceptable if this analysis demonstrated that personnel could access the vital areas, and take necessary actions to mitigate the consequences of the accident, without exceeding the dose criteria of GDC 19, "Control Room."

2.4.2 Technical Evaluation

In Attachment E of the October 2008 submittal, the licensee states that LSCS has three vital areas as defined in NUREG-0737 II.B.2. These are the Control Room, the Technical Support Center (TSC), and the AEER where the remote shutdown panels are located.

In addressing the direct radiation exposures from the liquid contained in the ECCS system piping, the licensee notes that the current LSCS design basis analysis (based on the source term assumptions of Technical Information Document (TID) -14844, NUREG-0737, II.B.2.) are conservative over the 30-day duration of these analyses. Therefore, the existing analyzed doses remain conservative and are not reanalyzed. This is consistent with the NRC position contained in the memorandum from J. E. Rosenthal to A. C. Thadani, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" dated April 30, 2001 (ADAMS Accession No. ML011210348). The existing TID-14844-based evaluation of doses from ECCS piping shine is slightly conservative for AST because of AST delay in release of certain isotopes from the reactor, and the reduction in total iodine core inventory release fractions from 50 percent to 30 percent. This is offset to a degree by the AST increase in assumed core releases to greater than 1 percent for the cesium, tellurium, strontium and barium isotopic groups. The current design basis post accident vital area assessment is provided in Section 12.3.2, "Shielding," of the LSCS Updated Final Safety Analysis Report (UFSAR) Rev. 17. Table 12.3-7, "Post-Accident Dose Rates From Pipe contained Sources," of the UFSAR indicates that average 30 day whole body dose rates for the control room, the TSC and the AEER, are less than 0.001 rem/hr. This is well within the NUREG-0737 II.B.2 acceptance criteria for areas requiring continuous occupancy of 0.015 rem/hr.

Attachment E of the licensee's submittal does provide dose calculations for the TSC and AEER from gaseous airborne radioactivity based on the AST release assumptions. The licensee has calculated the mission dose an operator accessing the AEER of 2.07 rem TEDE. This AEER dose determination is based on an assumed 30 minute mission duration at the point in time during the accident when dose would be maximized, without any credit for respiratory protection. Based on a 30 day continuous occupancy, the licensee calculated the dose to an operator in the TSC of 1.02 rem TEDE.

Assuming a conservative bounding dose from systems containing a liquid accident source term of 0.72 rem (based on 0.001 rem/hr, 24 hr/day, and 30 day duration), the total accident dose for the TSC and AEER would be less than 1.74 rem TEDE and 2.79 rem TEDE respectively. This is well within the NUREG 0737 II.B.2 requirement of meeting the GDC 19 criteria (5 rem TEDE).

2.4.3 Conclusion

The NRC staff concludes that the licensee has demonstrated through calculations that the plant design will allow operator access to the TSC and AEER during accident conditions per the requirements of NUREG-0737 Action Plan Item II.B.2.

2.5 Electrical Engineering

2.5.1 Regulatory Evaluation

The following requirements and guidance documents are applicable to the NRC staff's review for Section 2.5 of this safety evaluation:

Appendix A of 10 CFR Part 50, GDC 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of SSCs that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single-failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

Appendix A of 10 CFR Part 50, GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

The regulation at 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that the safety-related electrical equipment which are relied upon to remain functional during and following design-basis events be qualified for accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

The regulations at 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," require that preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

The regulations at 10 CFR 50.67, "Accident Source Term," provides an optional provision for licensees to revise the AST used in design-basis radiological analyses.

RG 1.183 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 guide

establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This RG states that licensees may use the AST or TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," assumptions for performing the required environmental qualification analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID 14844) on environmental qualification doses.

RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

2.5.2 Technical Evaluation

The NRC staff has reviewed the electrical and environmental qualification portions of the license amendment request and provides the following evaluation:

The licensee has proposed using an AST to determine accident offsite and control room doses. The licensee stated that the AST involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to accident dose analyses.

The standby ac power system consists of five emergency diesel generator (EDG) sets that support the reactor units. Loads important to plant safety are divided into three divisions and are fed from redundant Class 1E engineered safety feature (ESF) switchgear groups. The Division 1 EDG set supports certain ESF loads and is common to Unit 1 and Unit 2 (i.e., swing diesel). Two Division 2 EDG sets are provided for other ESF loads, one each for Unit 1 and Unit 2. Individual Division 3 EDG sets are also provided for Unit 1 and Unit 2, but are only designed for use to support the High Pressure Core Spray (HPCS) system.

The EDGs are rated at 2600 kilowatt (kW) continuous and 2860 kW for 2000 hours. The NRC staff requested additional information on whether any loads were being added to the LSCS EDGs and if so, how the additional loads would affect the capability, capacity, and load sequencing of the EDGs. The licensee, in its November 18, 2009, response, stated that the SLC system pumps will be manually loaded onto its ESF bus, and that there are no additional loads that are automatically sequenced on to the EDGs. The licensee calculated the loading on the Division 2 EDGs, including the SLC system pump, is 2528.8 kW for Unit1 and 2471.8 kW for Unit 2, which is below the continuous rating of the EDGs. The loading on the Division 1 EDG is 2636.9 kW for an accident in Unit 1, and 2635.5 kW for an accident in Unit 2, which is marginally above the continuous rating, but below the 2000 hour rating of the Division 1 EDG.

The NRC staff requested additional information on changes to the EDG surveillance testing as a result of exceeding the continuous rating of the Division 1 EDG for postulated accident loading. The licensee, in its March 29, 2010, response, stated that a 24-hour endurance test is performed every 24 months on each EDG in accordance with SR 3.8.1.14. This SR demonstrates that the EDGs can start and run continuously near full load capability for an interval of at least 24 hours. Specifically, SR 3.8.1.14 states:

Verify each required diesel generator (DG) operating within the power factor limit operates for > 24 hours:

a. For ≥ 2 hours loaded at ≥ 2860 kW (110 percent of continuous rating); and
b. For the remaining hours of the test loaded ≥ 2400 kW and ≤ 2600 kW.

The testing required by SR 3.8.1.14 demonstrates that each EDG can operate at 110 percent of the continuous rating for a minimum of 120 minutes.

A maximum time of 125 minutes is required for full injection of the contents of the SLC tank. Because the postulated EDG loading of 2636.9 kW (101.4 percent of continuous rating) for a maximum duration of 125 minutes is well below the 2000 hr rating (2860kW) of the EDG, and because the surveillance test loads the EDGs to at least 110 percent of continuous rating for 120 minutes, the NRC staff finds that the postulated total EDG loading is bounded by the EDG's capability and capacity, and is, therefore, acceptable.

The NRC staff requested additional information on how the EDG 7-day fuel supply requirement is satisfied as a result of this increased loading on the EDGs. The licensee, in its August 3, 2010, letter, stated that the increase in EDG fuel consumption to supply the SLC system pumps was 3 gallons per hour for a total of just over 6 gallons for the 125 minutes required for full injection of the SLC tank. Each EDG requires 31,841 gallons of fuel for 7 days of operation with a minimum of 32,200 gallons in storage, which provides sufficient margin to cover the increase due to this AST amendment. The NRC staff reviewed the licensee's calculations and finds the minimum amount of stored fuel oil to be acceptable given the proposed changes.

The NRC staff requested the licensee to provide a list and descriptions of components added to its 10 CFR 50.49 program due to the AST, and additionally, confirm that these components are qualified for the postulated environmental conditions during and post accident. In its November 18, 2009, letter, the licensee stated that no components were added to the LSCS 10 CFR 50.49 program as a result of the AST adoption.

The NRC staff requested additional information regarding how the operators will be notified in the event that the SLC system becomes inoperable (e.g., control room annunciators). In its November 18, 2009, letter, the licensee stated that there are several control room alarms to alert operators of SLC system problems. They include alarms for SLC system tank level and temperature, injection valve control circuit function, and SLC pump protective trip indication. Based on review of the LAR and the response to the RAI discussed above, NRC staff finds that these alarms provide adequate indication to the operators of SLC system status.

The NRC staff requested additional information regarding how the SLC system meets single-failure criteria. In its November 18, 2009, letter, the licensee stated that the SLC system has redundant and electrically-independent logic systems, pumps and valves such that a single failure of one train will not adversely impact the redundant SLC train. The NRC staff independently reviewed the information contained in the licensee's LAR and RAI response and determined that the SLC system design satisfied single-failure criteria.

The NRC staff requested additional information regarding the ability of the SLC system to perform its required functions during accident conditions and in potentially high radiation areas.

In its November 18, 2009, letter, the licensee stated that the SLC system is installed in a mild environment and is not qualified to 10 CFR 50.49 requirements, though high radiation is expected in the area post-accident. The staff requested additional information regarding why the components subjected to high radiation during an accident were not included in the 10 CFR 50.49 program. In its March 29, 2010, letter, the licensee further clarified that the SLC system equipment will have completed its required function within the first 4 hours after the onset of the accident, and that the environment during the period the SLC equipment is required to operate is not significantly more severe than its normal operating environment. The NRC staff reviewed the time the SLC system is required to be functional, and the environmental conditions resulting from a LOCA during the time the equipment is required to operate, and concluded that the SLC equipment meets the mild environment criteria as defined in 10 CFR 50.49, paragraph (c), and therefore, does not need to be included in the scope of the environmental qualification program.

The NRC staff also reviewed the environmental qualification portion of the license amendment request. The licensee used the methodology contained in TID 14844 to determine the radiation doses in the existing environmental qualification analyses as stated in Attachment 1, Section 3.5 of the licensee's October 23, 2008 letter. As mentioned previously, the use of this methodology is consistent with the guidance contained in RG 1.183. Since the licensee will continue to use the TID 14844 methodology and no new equipment is added to its 10 CFR 50.49 program, the environmental qualification of equipment should remain bounding during full-scope implementation of an AST.

2.5.3 Conclusion

Based on the above evaluation, the NRC staff concludes that the proposed changes are in accordance with 10 CFR 50.49, 10 CFR 50.65, 10 CFR 50.67, the requirements of GDCs 17 and 18, and are consistent with the guidance in RGs 1.183 and 1.75. Therefore, the staff finds the proposed changes acceptable.

2.6 Radiological Consequences Analysis

2.6.1 Regulatory Evaluation

The following requirements and guidance documents are applicable to the NRC staff's review for Section 2.6 of this safety evaluation:

The NRC staff reviewed the licensee's evaluation of the radiological consequences of affected DBAs for implementation of the AST methodology, and the associated changes to the TS proposed by the licensee, against the requirements specified in 10 CFR 50.67(b)(2). The regulation requires that the licensee's analysis demonstrates with reasonable assurance that:

- An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 roentgen equivalent man (rem) TEDE.

- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage, would not receive a radiation dose in excess of 25 rem TEDE.

- 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 5 rem TEDE for the duration of the accident.

This SE addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results.

The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, the accident specific guideline values in Regulatory Position 4.4 of RG 1.183 and Table 1 of SRP Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulations, regulatory guides, and standards:

- 10 CFR Part 50.67;
- 10 CFR Part 50, GDC 19;
- RG 1.23, "Onsite Meteorological Programs," Rev. 1, March 2007;
- RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972;
- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3, June 2001;
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, November 1982;
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Rev. 0, July 2000;
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Rev. 0, June 2003;
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Rev. 0, May 2003;
- NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985;
- NUREG-0800, "Standard Review Plan," Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases," Rev. 3, March 2007;
- NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability Systems," Rev. 3, March 2007;

- NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Rev. 4, March 2007;
- NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000;
- NUREG-0800, "Standard Review Plan," Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Rev. 2, July 1981;
- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants;" and
- NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992.

The NRC staff also considered relevant information in the LSCS Units 1 and 2 UFSAR and TSs.

The DBA dose consequence analyses evaluated the integrated TEDE dose at the exclusion area boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) and the integrated dose to a LSCS Units 1 and 2 CR operator were evaluated for the duration of the accident. In addition, the TEDE dose to personnel in the TSC was also evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the "RADTRAD: Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03, computer code. NRC sponsored the development of the RADTRAD radiological consequence computer code, as described in NUREG/CR-6604. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff uses the RADTRAD computer code to perform independent confirmatory dose evaluations as needed to ensure a thorough understanding of the licensee's methods. Although the NRC staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses.

2.6.2 Technical Evaluation

2.6.2.1 Atmospheric Dispersion Estimates

Meteorological Data

The licensee initially used 6 years of onsite hourly meteorological data collected during calendar years 1998 through 2003, to generate new atmospheric dispersion factors (X/Q values) which were provided in an attachment to the October 23, 2008, AST license amendment request (LAR). The information was provided in the form of hourly meteorological data formatted for input into the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and joint wind speed, wind direction and atmospheric stability frequency distributions (JFDs) for input to the PAVAN atmospheric dispersion computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radiological Materials from Nuclear Power

Stations"). Because postulated release locations were identified for both elevated and ground level sources, wind speed and wind direction measurements at heights of 10.1, 60.0, and 114.3 meters above ground level were provided and used to calculate X/Q values for input into the dose assessment. Atmospheric stability categorization was based on temperature difference measurements (delta-T) between the 60.0 and 10.1 meter and 114.3 and 10.1 meter levels.

NRC staff performed a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. NRC staff noted that there appeared to be a relatively low frequency of reported unstable conditions. This was particularly noticeable when compared with historic data in the LSCS UFSAR. In addition, staff noted some differences when comparing the LSCS meteorological data with similar data from two other nuclear power plant sites located in northern Illinois, the Dresden Nuclear Power Station, Units 2 and 3 (Dresden) and Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities) sites. For example, wind speed and atmospheric stability data for 1995 through 1998, from the Dresden and Quad Cities sites in eastern and western northern Illinois, respectively, seemed more similar to each other than the 1998 through 2003 meteorological data from LSCS which is located between these two sites in central northern Illinois.

In response to an NRC request for additional information, the licensee provided supplemental information by letters dated September 28, 2009, and March 29, 2010, respectively. The licensee stated that a single meteorological contractor has reviewed meteorological data for the LSCS, Dresden and Quad Cities sites on a daily basis and found that differences between LSCS and the other two sites were routine, including for wind speed and atmospheric stability data. The LSCS meteorological tower is located at a site with fewer obstructions to wind flow, which leads to higher average wind speeds at LSCS. The licensee further noted that wind speeds recorded at LSCS are also much higher than at any of the other 18 nuclear-related facilities for which the contractor provides meteorological services. In the supplements, Exelon proposed several changes to the calculation of the X/Q values provided in the October 23, 2008, LAR, including use of only meteorological data from 1999 through 2003, to generate the X/Q values. The licensee removed meteorological data from the entire year of 1998 from all evaluations, as the data from January through July 1998, were from a different tower location than for the later data measured from 1999 through 2003.

With regard to further review of the 1999 through 2003, meteorological data and atmospheric stability measurements, examination of the data revealed that stable and neutral atmospheric conditions were consistently reported to occur at night, and unstable and neutral conditions during the day, as expected. Wind speed distributions for each measurement channel were quite similar from year to year. Wind direction frequency occurrence was reasonably similar from year to year at each level and among the three levels at the LSCS site. The combined data recovery of the wind speed, wind direction, and stability data was in the upper 90 percentiles at all three levels throughout the five year period. This meets the data recovery recommendation of RG 1.23, Revision 0, "Onsite Meteorological Programs."

The NRC staff also generated joint wind speed, wind direction, and atmospheric stability frequency distributions using the data supplied in the October 23, 2008, application to calculate EAB and LPZ X/Q values for comparison with X/Q values calculated by the licensee. A

comparison of JFDs of the ARCON96 data as compiled by the NRC staff with the JFDs used by the licensee as input to PAVAN showed reasonably good agreement.

The 1999 through 2003, meteorological data were used to generate CR, TSC, EAB, and outer boundary of the LPZ X/Q values for the LOCA and the FHA dose assessments evaluated in the current LAR. The NRC staff reviewed the available information relative to the onsite meteorological measurements program and the 1999 through 2003 ARCON96 meteorological data input files provided by the licensee. On the basis of this review, the NRC staff has concluded that this data provides an acceptable basis for making estimates of atmospheric dispersion for DBA assessments.

Control Room and Technical Support Center Atmospheric Dispersion Factors

To assess CR post-accident atmospheric dispersion conditions for the LOCA and FHA, the licensee generated X/Q values using the ARCON96 computer code and guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," RG 1.194, states that ARCON96 is an acceptable methodology for assessing CR X/Q values for use in design basis accident radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of this LAR for LSCS. Meteorological data measured from 1999 through 2003, were used to generate the X/Q values used in the dose assessment.

Although the licensee had postulated releases from several other locations in the initial LAR, in the revision provided by letter dated March 29, 2010, Exelon removed derivations of X/Q values not associated with the stack or MSIV release locations, as those X/Q values were no longer limiting based upon the reanalysis performed by the licensee. As a result, postulated release locations with respect to calculation of CR and TSC X/Q values resulting from an assumed LOCA or FHA were reduced to elevated releases from the stack and ground level releases from the base of the stack and the MSIVs. The base of the stack was determined to be the worst-case analysis location.

The licensee modeled the stack as an elevated release, which is consistent with the original LSCS licensing as documented in the original LSCS safety evaluation report, although the LSCS stack height of 112.8 meters above grade is less than 2.5 times the height of the highest adjacent building. Control room X/Q values were calculated using the ARCON96 and PAVAN atmospheric dispersion computer codes in accordance with RG 1.194 guidance for elevated releases. For the 0- to 2-hour time period, the licensee compared the maximum PAVAN X/Q value with the ARCON96 value and determined that the PAVAN derivation resulted in the more conservative X/Q value. The licensee noted that the higher of the maximum sector and site limit X/Q value for the actual distance and for distances nearer to and further from the CR intake for each time period was used in the assessment. The 2- to 8-hour and 8- to 24-hour time periods only used results from the ARCON96 calculations. For both the 1- to 4-day and 4- to 30-day time periods, the licensee used the effective X/Q value as allowed for stack releases per RG 1.194. This deterministic approach assumed that the stack plume reversed direction one hour daily throughout the event. Both the maximum PAVAN value and ARCON96 X/Q value were used in this analysis as outlined in Equations [1] and [2] of RG 1.194. While the actual horizontal distance between the stack and CR intakes is 54 meters, with respect to postulated

releases from the stack to the CR intake and TSC, the licensee generated X/Q values using PAVAN for distances ranging from 50 to 5000 meters to find the limiting value. The licensee also generated X/Q values for distances ranging from 100 to 4500 meters from the stack to the TSC although the actual separation distance is 165 meters. When making calculations using the PAVAN computer code, the licensee adjusted the input height of release to reflect the actual vertical separation between the top of the stack and the CR intake and TSC receptor locations.

For postulated releases from the base of the stack and the MSIVs, the licensee assumed a ground level point source release and calculated X/Q values using the ARCON96 computer code and the 10.1 meter wind data and the 61.0 and 10.1 meter delta-T data. The shortest horizontal distance between each release location and the CR intake or TSC was used as the distance between the release and receptor locations.

The licensee stated that the generated X/Q values model the limiting doses and that all potential release scenarios were considered, including those due to loss of offsite power or other single failures. The licensee reviewed pertinent drawings and performed site walkdowns to ensure that there is no more limiting release pathway. Further, the licensee stated that use of the CR intake X/Q values for input into the dose assessment for unfiltered in-leakage is conservative in that the intakes are closer to the significant release sources than the CR itself or the negative pressure portions of the ductwork.

In summary, the NRC staff reviewed the licensee's assessments of CR and TSC post-accident atmospheric dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The staff qualitatively reviewed inputs to the ARCON96 and PAVAN computer runs for the CR and TSC X/Q value assessment and found them generally consistent with site configuration drawings and staff practice. In addition, the staff performed a check of the licensee's atmospheric dispersion estimates by running the ARCON96 and PAVAN computer codes with application of the RG 1.194 criteria and obtained similar results for a sample of cases. On the basis of this review, the NRC staff has concluded that the licensee's CR and TSC X/Q values listed in Table 2.1-1 are acceptable for use in the DBA CR dose assessments.

Offsite Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ X/Q values for the LOCA and FHA events using guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the PAVAN atmospheric dispersion computer code. Meteorological data measured from 1999 through 2003 were used to generate the X/Q values used in the dose assessment. The data were divided into a relatively large number of wind speed categories at the lower wind speeds to generate the JFD input file. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," states that JFDs used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results.

Postulated discharges from the 112.8 meter stack were treated as elevated releases. The EAB and LPZ X/Q values were calculated using the 114.3 meter wind data and the 114.3 and 10.1 meter delta-T data at distances of 509 meters and 6400 meters, respectively. However, as stated in RG 1.145, for stack releases, the maximum ground-level concentration in a sector may occur beyond the EAB distance. Therefore, for stack releases, X/Q calculations should be

made in each sector at each minimum boundary distance and at various distances beyond the EAB distance to determine the maximum relative concentration. To facilitate this recommendation the licensee generated EAB X/Q values for a range of distances between 520 and 3300 meters. The licensee assumed a terrain height of 17 meters in the south southwest through northwest sectors at and beyond distances of 1600 meters. This is the maximum terrain height in the LSCS site area. The licensee also assumed fumigation conditions for a half-hour time period as recommended in RG 1.145 for inland sites.

For postulated releases from the turbine building, the licensee assumed a ground level release and calculated X/Q values using the 10.1 meter wind data and the 61.0 and 10.1 meter delta-T data. The licensee made multiple calculations to confirm that the limiting case was at the actual EAB distance of 423 meters. The licensee also assumed a building minimum cross-sectional area of 2205 square meters and a containment height of 56.1 meters as inputs to the calculations.

In summary, the NRC staff qualitatively reviewed the inputs to the PAVAN computer runs and found them generally consistent with site configuration drawings and staff practice. In addition, staff reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting EAB and LPZ X/Q values are presented in Table 2.1-2. On the basis of this review, the staff has concluded that these X/Q values are acceptable for use in DBA EAB and LPZ dose assessments.

Secondary Containment Drawdown-Meteorology

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states that the effect of high winds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the data set. The licensee estimated the relevant LSCS 95 percent wind speed as about 28.2 miles per hour at the 61 meter measurement level. The NRC staff confirmed this estimate from the 1998 through 2003 onsite wind data.

2.6.2.2 Radiological Consequences of Design-Basis Accidents

The licensee has proposed a licensing basis change for its offsite and control room DBA dose consequence analysis for LSCS. The proposed change will implement an AST methodology for determining DBA offsite and CR doses. For full implementation of the AST DBA analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11 and GDC 19. To incorporate a full implementation of the AST, RG 1.183 Position 1.2.1 specifies that the DBA LOCA must be reanalyzed.

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS reactor or secondary coolant specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences of the following DBAs:

- Loss-of-Coolant Accident (LOCA)
- Fuel-Handling Accident (FHA)

Although the current licensed maximum reactor core power level for LSCS is 3489 MWt, the above analyses assume a maximum core power of 3559 MWt which includes 2 percent uncertainty.

2.6.2.2.1 Loss of Coolant Accident

A LOCA is a failure of the reactor coolant system (RCS) that results in the loss of reactor coolant which, if not mitigated, could result in fuel damage, including a core melt. During a LOCA, the primary coolant blows down through a break, depressurizing the RCS. As the pressure builds in the drywell, steam and other gases expand into the wetwell. While passing through the suppression pool water, the steam is condensed, thereby reducing the pressure in the wetwell and drywell. A reactor trip occurs and the ECCS actuates to remove fuel decay heat. Thermodynamic analyses, performed using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis assumes that ECCS is not effective and that substantial fuel damage occurs. Appendix A of RG 1.183 identifies acceptable radiological analysis assumptions for a LOCA. The source term and release pathways related to the LOCA are discussed below.

Source Term

The licensee projected the core inventory of fission products using the ORIGEN 2.1 isotope generation and depletion computer code. The source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD), to achieve equilibrium) conditions and the worst-case inventory for the selected isotopes were used for the core inventory. The fission product inventory is based on a 2-year fuel cycle with a nominal cycle of 711 EFPD. The ORIGEN 2.1 computer code is acceptable to the staff for estimating the core inventory, as discussed in RG 1.183. The standard 60-isotope RADTRAD inventory file was used in the licensee's accident dose calculations for airborne radioactivity.

Fission products from the damaged fuel are released into RCS and then into the primary containment. It is anticipated that the initial release to the drywell will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid at the start of the accident. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The gap inventory release phase begins two minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. This phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases. The inventory in each release phase is released at a constant rate starting at the onset of the phase and continuing over the duration of the phase.

LOCA Fission Product Transport

The LOCA considered in this evaluation is a complete and instantaneous severance of one of the recirculation loops. The pipe break results in a blowdown of the RPV liquid and steam to the drywell via the severed recirculation pipe. The resulting pressure buildup drives the mixture of steam, water, and other gases through the suppression pool water and into the primary containment. The suppression pool water condenses the steam and reduces the pressure. After the initial RPV blowdown, ECCS water injected into the RPV will spill into the drywell, transporting fission products to the suppression pool and then into the primary containment.

The licensee has conservatively assumed that the fission product release from the reactor is to mix instantaneously and homogeneously throughout the drywell. No mixing between the drywell and the wetwell is assumed for the first 2 hours. The licensee assumed that the initial blowdown occurs before fuel damage commences, and that the AST source terms are based on a non-mechanistic loss of ECCS flow to the reactor for two hours. After ECCS flow restoration, the rapid steaming of the ECCS liquids is assumed to quickly displace significant fractions of the airborne activity in the drywell through downcomers into the suppression chamber, providing the mixing mechanism. Conservatively, the licensee did not credit any reduction by suppression pool scrubbing for fission products transferred to the primary containment through the suppression pool. Therefore, after 2 hours, complete mixing of activity in the drywell volume to the suppression chamber airspace is assumed. The NRC staff reviewed the licensee's assumptions and, based on engineering judgment, finds that the licensee's assumptions regarding drywell and containment mixing provide a conservative estimate of the fission product release and is, therefore, acceptable.

The licensee assumes that a portion of the fission products released from the RPV will plateout in the drywell and primary containment due to natural deposition processes. The licensee models this deposition using the 10th-percentile values in the model described in the staff-accepted NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (i.e., the "Powers Model"). The licensee did not assume natural deposition of elemental or organic forms of iodine in the drywell or containment. The licensee's assumptions on drywell/containment mixing and natural deposition processes are consistent with the guidance in RG 1.183.

The AST assumes that the iodine released to the containment consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption in this iodine speciation is predicated on maintaining the containment sump water above a pH of 7.0. The licensee proposes to use the SLC system to inject sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over into the suppression pool. Sodium pentaborate, a base, will neutralize acids generated in the post-accident primary containment environment. The control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged. As such, the new role being assigned to the SLC is a safety-related role, and was discussed more fully in Section 2.1.2.2 and 2.2.

LOCA Release Pathways

The release to the environment is assumed to occur through the following pathways:

- Primary containment leakage;

- MSIV leakage; and
- ECCS leakage.

Under the previously used TID-14844 source term assumption of instantaneous core damage and fission product release, the initial blowdown would also include all of the released fission products, a fraction of which would be retained by the suppression pool water. Under the AST, a substantial fraction of the fission product release from the core occurs after the initial blowdown is complete. Therefore, the licensee did not credit any reduction in fission products transferred to the wetwell air space by suppression pool scrubbing, assuming instead, a well-mixed wetwell air space and drywell after 2 hours.

Primary Containment Leakage Pathway

In its October 23, 2008, submittal, the licensee proposes to change TS 5.5.13, "Primary Containment Leakage Rate Testing Program." The proposed change will increase the maximum allowable primary containment leakage rate, L_a, at P_a, from 0.635 percent to 1.0 percent of primary containment air weight per day. For the LOCA AST analysis, the primary containment is projected to leak at the proposed L_a of 1.0 percent of its contents by weight per day for the entire 30-day accident duration. The NRC staff reviewed the LOCA AST analysis using the proposed L_a of 1.0 percent of its contents by weight per day. The staff finds that the resulting doses are below the regulatory limits as stated in 10 CFR 50.67. Therefore, the proposed change is acceptable.

Leakage from the primary containment collects in the free volume of the secondary containment and is subsequently released to the environment via ventilation system exhaust. Two minutes after a LOCA and gap release, the standby gas treatment system (SGTS) fans start and drawdown of the secondary containment commences. Seventeen minutes after the LOCA, the SGTS has established a negative pressure with respect to the environment in the secondary containment and credit for filtration through the SGTS is taken. All releases from the reactor enclosure to the environment are modeled as ground-level releases. The NRC staff finds that licensee's assumptions for the primary containment leakage are in accordance with the guidance in RG 1.183, and are, therefore, acceptable.

MSIV Leakage Pathway

A source of containment leakage that bypasses the secondary containment is MSIV leakage. The four main steamlines, which penetrate the primary containment, are automatically isolated by the MSIVs in the event of a LOCA. There are two MSIVs on each steam line, one inside the drywell (i.e., inboard) and one outside the primary containment, (i.e., outboard). The MSIVs are functionally part of the primary containment boundary and design leakage through these valves provides a leakage path for fission products to bypass the secondary containment and enter the environment as a ground level release.

In its October 23, 2008, submittal, the licensee proposes to revise, TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)." The proposed change revises SR 3.6.1.3.10 to increase the leakage limit through any one main steamline to 200 scfh. Currently, the SR requires verification that the leakage rate through any one main steamline is less than or equal to 100 standard cubic feet per hour (scfh), and that the leakage rate through all four main steamlines is less than or equal to 400 scfh, when tested at greater than or equal to 25.0 psig.

The proposed change increases the leakage limit through any one main steamline from 100 scfh to 200 scfh. The combined leakage rate limit through all four main steamlines is not being changed and will continue to apply at test pressures greater than or equal to 25 psig. Both the 200 scfh leakage rate through any one main steamline and the 400 scfh leakage rate through all four main steamlines are used in the LOCA AST analysis.

The licensee assumes that the outboard MSIVs fail to close on all four main steamlines with one line broken upstream of the inboard MSIV. The licensee assumes a maximum MSIV leakage of 200 scfh in the broken line, one of the unbroken lines is assumed to leak at 200 scfh, and the other two lines are assumed not to leak. These leakrates are consistent with the proposed increase in the SR MSIV leakage criterion. No reduction in leakage is assumed at 24 hours, for conservatism.

The licensee conservatively assumes that the fission products released from the core are dispersed equally throughout the drywell. Following the initial blowdown of the RPV, the fuel heats up, fuel melt begins, and steaming in the RPV carries fission products to the drywell. When core cooling is restored, steam is rapidly generated in the core. This steam and the ECCS flow carry fission products from the core to the primary containment, resulting in well-mixed RPV dome and primary containment fission product concentrations. Once the rapid steaming stops, the primary containment contents can flow back through a severed main steamline (conservatively assumed in lieu of the recirculation line break for this release pathway only) and would be available for release via leakage through the MSIVs.

The licensee's analysis does not take credit for the MSIV leakage control system but does propose to take credit for aerosol and iodine removal in the mainsteam lines. The licensee iodine removal modeling assumes well-mixed control volumes. Only the volumes associated with horizontal runs of seismically qualified main steamline piping are included in the modeling of iodine aerosol deposition. The licensee assumes two aerosol settling volumes (nodes) for the unbroken main steamline; one node between the RPV and the inboard MSIV, and the other node between the inboard MSIV and the turbine/auxiliary building (secondary containment) wall. The main steamline conservatively assumed to be broken does not have the volume between the RPV and inboard MSIV available for iodine removal, so only assumes one aerosol settling node. The licensee's main steamline modeling is conservative because it minimizes aerosol deposition credit.

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

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

The NRC staff acknowledges that aerosol settling is expected to occur in the main steamline piping, but because of recent concerns regarding the AEB-98-03 report and the lack of further information, the staff does not know how much deposition (i.e., which settling velocity value) is appropriate. The licensee used a model based on the methodology in AEB-98-03, but included some additional conservatism to address the NRC staff's concerns about the applicability of the AEB-98-03 methodology to LSCS. The 10th percentile aerosol settling velocity is a smaller value (and estimates less aerosol settling) than 90 percent of the calculated settling velocities in AEB-98-03. Based upon AEB-98-03, use of the 10th percentile settling velocity is more conservative than the use of the median settling velocity noted as reasonable in AEB-98-03. Given the aforementioned conservatism and the presence of a seismically qualified condenser, the NRC staff finds the LSCS main steamline aerosol settling model to be reasonable and appropriate.

The licensee also assumed deposition of elemental iodine in the main steamline piping. The licensee used the model described in a letter report dated March 26, 1991, by J. E. Cline, "MSIV Leakage Iodine Transport Analysis," (hereafter, the Cline report). The Cline report provides elemental iodine deposition velocities, re-suspension rates and fixation rates. The deposition velocities were used in the well-mixed model formulation described above for use with AEB-98-03. Because elemental deposition is not gravity dependent, the licensee assumed elemental iodine deposition occurs on the entire surface area of the horizontal and vertical piping. The licensee evaluated the effects of re-suspension, as described in the Cline report, and found the dose impacts to be small. Any re-suspended iodine was modeled as organic iodine and assumed released instantly. No removal of organic iodine was assumed by the licensee. RG 1.183 refers to the Cline report as a source of guidance on gaseous iodine transport modeling that is acceptable to the NRC staff. The NRC staff has determined that the licensee's application of the Cline report assumptions is acceptable because it is conservative and consistent with the guidance of RG 1.183.

ECCS Leakage Pathway

During the progression of a LOCA, some fission products released from the fuel will be carried to the suppression pool via spillage from the RCS. Post-LOCA, the suppression pool is a source of water for the ESF systems. Since portions of these systems are located outside the primary containment, potential leakage from these systems is evaluated as a radiation exposure pathway. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated leakage are analyzed and combined with the radiological consequences from other fission product release paths to determine the total calculated radiological consequences from the LOCA. ECCS components are located in the Reactor Building.

With the exception of noble gases, fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the suppression pool water at the time of release from the core. The total ECCS leakage from all components in the ECCS systems is assumed to be 5 gpm, which is assumed to start immediately after the onset of a LOCA. With the exception of iodine, remaining fission products in the recirculating liquid are assumed to be retained in the pool water. The licensee determined that the post-LOCA temperature of suppression pool water recirculated through the ECCS system is less than 212°F. In accordance with RG 1.183, the licensee, therefore assumed that 10 percent of the iodine activity in the leaked liquid becomes airborne. The reduction in ECCS leakage activity by dilution in the Reactor Building volume is not credited. The radioactive iodine that is postulated to be available for release to the environment due to ECCS leakage is assumed to be 97 percent elemental and 3 percent organic. The release continues for 30 days.

Two sources of potential ESF leakage were included in the release model. The first is ESF system leakage directly into secondary containment. The analysis assumes a value of 5 gallons per minute. The leakage value is more than two times the acceptance criteria for the sum of the simultaneous leakage from all components in the ESF recirculation systems. Leakage was assumed to start immediately after the onset of a LOCA. The NRC staff reviewed the licensee's analytical treatment of ESF leakage and found it to be consistent with the guidance of RG 1.183, and therefore, acceptable. The second source of potential ESF leakage is into the condensate storage tanks (CSTs). The licensee performed an analysis that concluded that the dose from this pathway is negligible. The NRC staff evaluated the effects of the leakage to the CSTs, and agreed with the licensee's assessment that the dose from this pathway will be negligible because iodine will not evolve from a basic water solution, and the CST solution will continue to remain basic with the in-leakage.

Control Room Doses

The LSCS CR envelope has historically been treated as consisting of the CR and the AEER, with a shared filtered emergency makeup system and separate filtered recirculation systems. In the AST LOCA analysis, standard continuous occupancy assumptions are applied to the CR. However, AEER occupancy is only required for the safety related action of starting the fan that provides containment air mixing as required per 10 CFR 50.44(c)(1) for combustible gas control. This mission is assumed to be performed by an operator not assigned full time to the CR, but dispatched from the CR. Although the total expected time for this mission outside of the CR is nine minutes, the licensee assumes 30 minutes for the AST LOCA analysis. The worst-case timing for this operation would be starting at time zero because of exposure to releases during reactor enclosure drawdown. No credit is taken for any filtration provided by the makeup filter or AEER recirculation filter system. Therefore, the features that control radioactivity in the AEER, such as filtered intake, filtered recirculation, and positive pressurization are not required for this mission.

The CR and AEER share a makeup filter system, but have separate recirculation filter systems. Nominally, 37.5 percent of the makeup flow is directed to the CR and 62.5 percent is directed to the AEER. Assuming a 10 percent reduction in flow to account for bounding tolerances (i.e., from 4000 cfm to 3600 cfm), CR flow would be 1350 cfm and AEER flow would be 2250 cfm for the nominal flow split. In the AST LOCA analysis, splits of 25 percent, 37.5 percent, and 50 percent to the CR are analyzed in this distribution with the balances directed to the AEER. The licensee's analyses demonstrate that the bounding doses occur with the minimum

25 percent flow to the CR. The bounding values for dose analysis purposes were used to demonstrate compliance with 10 CFR 50.67. The NRC staff finds the licensee's approach acceptable.

The CR/AEER makeup filter charcoal adsorber credit is based on 90 percent efficiency for elemental and organic iodines, rather than the historically credited 95 percent. Values were minimized for the AST analysis to provide additional margin. However, no changes to the TSs regarding filter efficiency are proposed. Because of the presence of high-efficiency particulate air (HEPA) filtration in the makeup filter train, aerosol removal efficiency is credited at 99 percent. No aerosol removal is credited in the CR or AEER recirculation filter trains.

Recirculation filter bypass for the CR is assumed to be 5 percent of the minimum CR supply flow. That is 900 cfm for the CR recirculation filter. A CR recirculation filtered in-leakage rate of 2400 cfm is assumed. The in-leakage value is 200 percent above the current design basis assumption used in the past, and conservatively well above actual in-leakage determined by tracer gas testing. The assumption provides operational margin to be managed under the LSCS Control Room Envelope Habitability Program. In addition to the filtered in-leakage and the 5 percent filter bypass, another 50 cfm of unfiltered in-leakage is assumed into the ductwork downstream of the CR recirculation filters and upstream of the supply fans for the CR. The allowance is based on historical estimates of maximum credible leakage and bounds the tracer gas results.

Direct Gamma Shine Doses

The licensee reviewed the TID-14844 based analyses of LSCS CR dose due to gamma shine from sources external to the CR. These analyses are summarized in LSCS UFSAR Table 6.4-2. The licensee determined which of the existing assessments will continue to be used and which would be reanalyzed with AST assumptions. The licensee determined that the following three contributors - reactor building plateout, control building intake filter loading, and exhaust cloud external to station - would be reanalyzed with AST assumptions:

(1) Reactor Building Refuel Floor, Wall and Ceiling Plateout Source

According to the licensee, plateout is much less for AST due to in-containment deposition and iodine's chemical form. Only elemental iodine is deposited on the walls and the ceiling. Aerosols will only experience gravitational settling to the floor, yielding no unshielded wall or roof shine toward control building roof.

The licensee modeled a primary containment (PC) leak, treating the refuel floor surface as a compartment, with output of detailing isotopic loading. The containment leakage and secondary enclosure exhaust is assumed to be proportional to the above and below the refuel floor free volumes. ECCS leakage is not modeled since that activity is expected to be confined below the refuel floor due to the location of related equipment.

The licensee formulated the elemental iodine wall deposition using SRP 6.5.2. This formula for plateout is generally applicable to wetted surfaces inside containment. The NRC staff finds it conservative to assume plateout at the rate calculated for a wetted surface, as it will over-predict the amount of actual plateout that will be seen on the dry reactor building surface. The calculated wall loading due to plateout as a function of time is linearly integrated over the

720-hour event duration. These time-integrated sources are then adjusted for CR occupancy credit in accordance with RG 1.183. To determine and analyze the resulting dose consequence from the surfaces of the reactor building wall and ceiling that can be "seen" from the CR through the control building roof and intervening floors, the licensee uses the same geometry derived in its original TID-14844 based analysis. Based on the above, the NRC staff finds that the licensee used conservative assumptions for the calculation of the gamma shine contribution from the reactor building floor. The NRC staff agrees with the licensee's modeling and finds the licensee's assessment of this streaming source acceptable.

(2) Control Building Intake (and Recirculation) Filter Loading Source

For the control building intake filter loading source, the licensee determined a conservative filter (system) loading. This loading reflects material contained in all intake and in-leakage streams, based on the assumption that aerosol, elemental, and organic iodine activity, even if initially unfiltered, could be collected on the recirculation filters. This included:

1. the total makeup flow at 4400 cfm (nominal +10 percent) and
2. the CR and AEER total filtered in-leakage of 5600 cfm (2400 cfm for CR and 3200 cfm for AEER)
3. the CR and AEER total unfiltered in-leakage of 100 cfm (50 for the CR and 50 for AEER)

All of this filterable radioactive material is assumed to be deposited on a single filter that is treated as a CR compartment with no out-leakage pathway. The PC leakage, ECCS leakage, and MSIV leakage output data for this CR filter are then linearly integrated over the 720-hour event duration. As for the reactor building plateout, these time-integrated sources are then adjusted for CR occupancy credit in accordance with the applicable regulatory guidance.

The control building makeup filters and the CR and AEER recirculation filters are located above the control room. The licensee modeled streaming using a simplified geometry; 2-foot diameter by 2-foot long cylinder at the floor elevation of recirculation filters. The exposed individual, at head height would be approximately 13 feet from the center of this cylinder. This is the conservatively credited separation and only the 2 foot thick concrete intervening floor is credited for shielding. The NRC staff agrees with the licensee's modeling and finds the licensee's assessment of this streaming source acceptable.

(3) Exhaust Cloud External to Station Source

For the AST, the exhaust cloud external to station source was reanalyzed with a higher containment leak rate, MSIV leakage, and ECCS leakage. The external cloud activity concentration is calculated from the licensee's simulation of the PC leakage, ECCS leakage, and MSIV leakage sources. The licensee then presents the resulting concentrations by isotopic activity as a function of time.

For each source, releases are adjusted for CR occupancy and divided by 3600 sec/hr to yield a time-integrated concentration. These concentrations are then summed over the full accident duration and all release paths. The full integrated concentration is analyzed with a 1000 meter thick slab geometry to model a cloud. The NRC staff agrees with the licensee's modeling and finds their assessment of this streaming source acceptable.

The licensee calculated the total gamma shine dose to CR operator from external sources to be 0.04 rem. The licensee added the gamma shine dose value to the CR dose calculated for the release pathways discussed above. The NRC staff finds this formulation to be conservative and acceptable.

Technical Support Center (TSC) Dose Consequence Assessment

TSC has a single filter unit that provides both intake and recirculation filtration. This unit is made up of a prefilter, HEPA filter, two 2-inch charcoal adsorber beds and a downstream HEPA. The AST TSC dose analysis is performed without credit for the above filtration system. The entire filter flow is assumed to be outside air at 1000 cfm, supplemented by 3000 cfm of also unfiltered in-leakage. The licensee shows that the TSC 30-day inhalation and immersion dose was thoroughly analyzed for the LOCA and each of the other DBAs. In examining their post-accident TSC dose consequences, the licensee finds that the 30-day doses do not exceed 5 rem TEDE. The NRC staff reviewed the licensee's assumptions and results and determined that the licensee followed the regulatory requirements of Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 and the regulatory guidance of NUREG-0737, "Clarification of TMI Action Plan Requirements." Therefore, the staff finds, based on engineering judgment, the licensee's examination of the DBA dose consequences to the TSC acceptable.

LOCA Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the staff are presented in Table 2.2-2. Based upon the information provided by the licensee, the NRC staff finds that the licensee used analysis, methods, and assumptions consistent with the guidance of RG 1.183, except were discussed and accepted above. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, which the licensee's estimates of the EAB, LPZ, CR, and TSC doses, as shown in Table 2.2-1, for the LOCA will continue to comply with these criteria.

2.6.2.2.2 Fuel-Handling Accident

The FHA assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto other fuel bundles in the core. The drop of a fuel assembly in the reactor well (vessel cavity) over the reactor core was found to be the limiting design basis case. A fuel assembly and mast is postulated to drop from the maximum height allowed by the refueling platform and to fall onto the fuel in the reactor. The reactor vessel head is assumed to be off. At this location, the maximum drop (free fall distance) is 34 feet for the fuel assembly and for the mast. The analysis assumes a water depth of 23 feet above the assemblies seated in the reactor pressure vessel.

The extent of damage for both cases is calculated based on the free fall distance and the resulting kinetic energy of the dropped assembly. For the revised AST FHA event, the licensee based its fraction of core fuel damage on the GESTAR II limiting case of damaging 172 fuel pins (based on a "Heavy Mast" design) from GE12 and GE14 10x10 fuel bundle arrays with the equivalent of 87.33 pins per bundle, and with all of the damaged fuel assumed to have a limiting

radial peaking factor of 1.7. A post-shutdown 24-hour decay period was used to determine the release activity inventory.

The licensee calculated the activity in the gap using the non-LOCA gap inventory fractions presented in Table 3 of RG 1.183. These release fractions for non-LOCA events are acceptable for light-water reactor fuel with a peak fuel exposure up to 62,000 MWD/MTU. The fuel peak exposure is acceptable provided that the fuel operating linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for fuel burnup or exposure exceeding 54,000 MWD/MTU in accordance with Table 3 of RG 1.183, Footnote 11. The licensee stated that the limits above will not be exceeded at LSCS and that these fuel limits will be evaluated before each refuel cycle.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Following the guidance in RG 1.183, Regulatory Position 3.5 and Appendix B, Section 1.3, the licensee assumed that the chemical form of radioactive iodine released from the fuel to the SFP consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine as part of the CsI re-evolves as elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

The fission product inventory in the core is largely contained in the fuel pellets that are enclosed in the fuel rod clad. However, the volatile constituents of this inventory will migrate from the pellets to the gap between the pellets and the fuel rod clad. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the pool water, depending on their physical and chemical form. The fission products released from the pool are assumed to be released to the environment over 2 hours. The licensee assumes the Reactor Building exhaust rate is set at 0.1 air changes per minute to assure an essentially complete release within 2 hours.

The licensee evaluated a worst-case (with respect to CR dose) scenario for the CRAF system utilization, corresponding to use of the +10 percent tolerance in the outside air intake rate during a 20-minute initial period of no filtration, followed by a corresponding filtered intake rate with the negative 10 percent tolerance. This maximizes the amount of unfiltered air and minimizes the amount of filtered air being circulated within the CR. All three CR to AEER flow splits (25 percent CR/75 percent AEER, 37.5 percent CR/62.5 percent AEER, and 50 percent CR/50 percent AEER) from the LOCA AST analysis are evaluated to ensure worst-case results. The other CRAF flow rates, filter efficiencies, and operational time delays from the LOCA AST analysis are utilized as well. The licensee's analyses demonstrate that the bounding doses occur with the minimum 50 percent flow to the CR. The bounding values for dose analysis purposes were used to demonstrate 10 CFR 50.67 compliance. Most significantly, this includes in-leakage rates totaling 2450 cfm. This is more than 200 percent of the historically assumed value. The NRC staff finds that the licensee's assumptions used to model the LSCS CR are conservative and maximize the analyzed post-accident dose to the CR.

FHA Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The assumptions found acceptable to the staff are presented in Table 2.2-3. Based upon the information provided by the licensee, the staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183, except were discussed and accepted above. The NRC staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The NRC staff concludes that, with reasonable assurance, the licensee's estimates of the EAB, LPZ, and control room doses, as shown in Table 2.2-1 for the FHA will continue to comply with these criteria.

2.6.3 Radiological Consequences - Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed license amendment at LSCS. The staff finds that analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above were used. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, CR, and TSC doses, as shown in Table 2.2-1 will continue to comply with these criteria. Therefore, the NRC staff finds the proposed license amendment acceptable with regard to the radiological consequences of postulated DBAs.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the LSCS design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria, expressed in terms of whole body, thyroid, and skin doses, are superseded by the TEDE criteria of 10 CFR 50.67, or fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the LSCS design basis.

2.6.4 Tables

Table 2.1-1
LaSalle Control Room Atmospheric Dispersion Factors (X/Q Values, sec/m^3)

Source to Receptor	Type of Release	0 -2 hrs	2 – 8 hrs	8 – 24 hrs	1 - 4 days	4 – 30 days
Stack to CR/AEER**	Elevated	1.17×10^{-5}	1.00×10^{-36}	1.00×10^{-36}	7.17×10^{-8}	2.25×10^{-8}
Stack toTSC	Elevated	3.13×10^{-6}	0.00*	0.00*	1.86×10^{-8}	5.75×10^{-9}
Stack to CR/AEER	Ground	6.83×10^{-4}	5.04×10^{-4}	2.13×10^{-4}	1.34×10^{-4}	9.70×10^{-5}
MSIV to CR/AEER	Ground	8.84×10^{-4}	6.70×10^{-4}	2.61×10^{-4}	1.67×10^{-4}	1.32×10^{-4}
MSIV to TSC	Ground	9.11×10^{-4}	6.39×10^{-4}	2.60×10^{-4}	1.54×10^{-4}	1.30×10^{-4}

* Value as calculated by the ARCON96 computer code is negligibly small
** Auxiliary Electric Equipment Room

Table 2.1-2
LaSalle EAB and LPZ Atmospheric Dispersion Factors (X/Q Values, sec/m^3)

		Elevated*	Ground Level
EAB	**0 – 0.5 hrs**	8.80×10^{-5}	-----
	0.5 – 2 hrs	2.74×10^{-6}	-----
	0 – 2 hrs	-----	6.63×10^{-4}
LPZ	**0 – 0.5 hrs**	1.05×10^{-5}	-----
	0.5 – 2 hrs	1.77×10^{-6}	-----
	0 – 2 hrs	-----	2.65×10^{-5}
	2 – 8 hrs	8.34×10^{-7}	1.08×10^{-5}
	8 – 24 hrs	5.72×10^{-7}	6.87×10^{-6}
	1 – 4 days	2.53×10^{-7}	2.63×10^{-6}
	4 – 30 days	7.81×10^{-8}	6.74×10^{-7}

* Maximum χ/Q values occurred beyond actual EAB distance at a minimum distance of 2500 meters

Table 2.2-1
LSCS Units 1 and 2
Calculated Radiological Consequences TEDE [1] (rem)

Design Basis Accident	EAB[2]	LPZ[3]	CR	TSC
Loss of Coolant Accident	2.59	0.27	4.27	1.2
Dose Criteria	25	25	5	5
Fuel Handling Accident	1.45	0.059	1.14	
Dose Criteria	6.3	6.3	5.0	

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone

Table 2.2-2
LSCS Units 1 and 2
Parameters and Assumptions for the LOCA

Parameter	Value
Core Power Level	3559 MWt
Core Source Terms	RG 1.183 based list of 60 Core Isotopes, with bounding activities
Dose Conversion Factors	FGR 11 and 12 for Inhalation CEDE and cloud submersion EDE.
Primary Containment Volume Drywell free volume Wetwell airspace volume	394,338 ft^3 229,538 ft^3 164,800 ft^3
Minimum Suppression Pool Water Volume	128,800 ft^3
Primary Containment Leak Rate	1.0% per day
Secondary containment volume	2.875×10^6 ft^3
Mixing in Secondary Containment Volume	None Credited
Secondary Containment Bypass	None except for MSIV leakage
Containment Activity Removal Mechanisms	Natural Deposition
SGTS flow rate	4,000 cfm
SGTS filter efficiency	99% (not including bypass effects)
Reactor Building Drawdown Time	15 minutes
MSIV Leakage Rates	400 scfh total 200 scfh single line
ECCS Leakage into Secondary Containment Leak Rate Fraction Flashed Filtered by SGTS	 5 gpm 10.0% Yes
Emergency makeup filter unit flow	4000 ± 10% cfm
CR recirculation filter flow	18,000 cfm
CR recirculation filter bypass	900 cfm
CR outside air unfiltered in-leakage after makeup filter	55.2 cfm

Table 2.2-2
LSCS Units 1 and 2
Parameters and Assumptions for the LOCA (continued)

Parameter	Value
CR outside air unfiltered in-leakage into low pressure ductwork before recirculation filter	2,400 cfm
CR outside air unfiltered in-leakage rate after recirculation filter	50 cfm
CR intake filter charcoal efficiency	90%
CR recirculation filter charcoal filter efficiency	70%
CR HVAC system activation times after LOCA signal makeup filter recirculation filter	 20 minutes 4 hours
CR occupancy requirements	0-24 hrs: 1.0 1-4 days: 0.6 4-30 days: 0.4
AEER occupancy	Non full-time operator has 30 minutes to start the Hydrogen Recombiner system fan for containment mixing
AEER outside air unfiltered in-leakage	100,000 cfm
AEER filtration system consideration	No credit for protection by the makeup filter or the AEER recirculation filter
AEER volume	68,800 ft^3
Atmospheric Dispersion Factors	Tables 2.1-1 and 2.1-2

Table 2.2-3
LSCS Units 1 and 2
Parameters and Assumptions for the FHA

Parameter	Value
Core Power Level	3559 MWt
Fuel assembly configuration and properties	10x10 in a 87.33 fuel pin bundle and 172 pins damaged
Radial Peaking Factor	1.7
Allowable fuel burnup and non-LOCA gap fractions	Table 3 of RG 1.183. Fuel burnup will not exceed 62 GWD/MTU. Linear heat generation rate (LHGR) for fuel >54 GWD/MTU will not exceed 6.3 KW/ft.
FHA radionuclide inventory	From Attachment A of AST design FHA analysis for the 60 isotopes forming the standard RADTRAD library, with decay to 24 hours.
Underwater Decontamination Factor	Noble Gases: 1 Particulate: infinity Iodine: 200
Dose conversion factors	Federal Guidance Reports 11 and 12
Secondary containment automatic isolation and filtration	Credited
Mitigation by CRAF system	Credited
Bounding CR fresh air intake	4000 ± 10% cfm
CR volume	117,400 ft^3
Reactor Building normal ventilation	Artificially set at an air change rate of 0.1 per minute
Atmospheric Dispersion Factors	Tables 2.1-1 and 2.1-2

3.0 COMMITMENTS

The following table identifies commitments made by EGC in its March 29, 2010, supplement:

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC (Yes/No)
Emergency Operating Procedure LGA-001, "RPV Control," will be revised to ensure that Standby Liquid Control (SLC) system injection is started from the boron solution storage tank during a design basis accident (DBA) loss-of-coolant accident (LOCA).	Upon Implementation	No	Yes
Emergency Operating Procedure LGA-001 will be revised to ensure no steps would terminate the injection during a DBA LOCA prior to emptying the SLC system boron solution storage tank (i.e., injection of the full content into the reactor pressure vessel).	Upon Implementation	No	Yes

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of the facilities components located within the restricted area as defined in 10 CFR Part 20 or a change in 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 that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (74 FR 15771; April 7, 2009). 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) 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.

7.0 REFERENCES

1. Letter from P.R. Simpson, Exelon Generation Company, LLC, to NRC Document Control Desk, "LaSalle County Station, Units 1 and 2, Facility Operating License Nos. NPF-11 and NPF-18, NRC Docket Nos. 50-373 and 50-374 - Additional Information Regarding Request for License Amendment Regarding Application of Alternative Source Term," October 23, 2008. (ADAMS Package No. ML083100149).

2. Letter from P.R. Simpson, Exelon Generation Company, LLC, to NRC Document Control Desk, "LaSalle County Station, Units 1 and 2, Facility Operating License Nos. NPF-11 and NPF-18, NRC Docket Nos. 50-373 and 50-374 - Additional Information Supporting Request for License Amendment Regarding Application of Alternative Source Term," September 28, 2009. (ADAMS Accession No. ML092710196).

3. General Electric Topical Report NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," August 1999, (ADAMS Accession No. ML993440253).

4. Letter from M.D. Lynch, NRC, to D.L. Farrar, Commonwealth Edison Company, "Issuance of Amendments [for LaSalle County Station] (TAC Nos. M93597 and M93598)," April 5, 1996, (ADAMS Accession No. ML021130117).

5. "Guidance on the Assessment of a BWR SLC System for pH Control," February 12, 2004, (ADAMS Accession No. ML040640364).

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Date: September 6, 2010

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

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 197 to NPF-11
2. Amendment No. 184 to NPF-18
3. Safety Evaluation

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RidsNrrDorlDpr Resource

ADAMS Accession No. ML101750625 NRR-058 *By Memo Dated

OFFICE	LPL3-2/PM	LPL3-2/LA	DSS/SCVB	DIRS/ITSB	DRA/AADB	DCI/CSGB
NAME	CGratton	THarris /CGratton for	RDennig*	RElliott	TTate*	RTaylor*
DATE	09/01/10	09/01/10	04/29 /10	07/09/10	06/28/10	10/07/09

DIRS/IHPB	DE/EEEB	DE/EMCB	OGC	LPL3-2/BC
UShoop*	GWilson*	MKhanna*	BHarris	RCarlson
05/20/10	06/17/10	12/10/09	08/26/10	09/06/10

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