August 23, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: APPLICATION OF ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NOS. MC2295 AND MC2296)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 185 to Facility Operating License No. NPF-39 and Amendment No. 146 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 27, 2004, as supplemented by letters dated October 25, 2004, October 10, 2005, April 27, 2006, May 30, 2006, June 16, 2006, and August 4, 2006.

The amendment revises the TSs to support application of an alternate source term methodology, with the exception that Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” will continue to be used as the radiation dose basis for equipment qualification.

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

Sincerely,

/RA/

Richard V. Guzman, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosures:
1. Amendment No. 185 to NPF-39
2. Amendment No. 146 to NPF-85
3. Safety Evaluation

cc w/encls: See next page

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Package Accession Number: **ML062210207**
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TS(s) Accession Number: **ML062350507**

OFFICE	LPL1-1/PM	LPL1-2/LA	SCVB/BC	AADB/BC	ITSB/BC	OGC	LPL1-2/BC (A)
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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. NPF-39

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

    A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 27, 2004, as supplemented by letters dated October 25, 2004, October 10, 2005, April 27, 2006, May 30, 2006, June 16, 2006, and August 4, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 185, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brooke D. Poole, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: August 23, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

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

Remove	Insert
1-2	1-2
1-6	1-6
1-7	1-7
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-16	3/4 3-16
3/4 3-31	3/4 3-31
3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
3/4 3-66	3/4 3-66
3/4 3-67	3/4 3-67
3/4 6-3	3/4 6-3
3/4 6-47	3/4 6-47
3/4 6-52	3/4 6-52
3/4 7-6	3/4 7-6
B 3/4 1-4	B 3/4 1-4
B 3/4 1-5	B 3/4 1-5
B 3/4 6-5	B 3/4 6-5
B 3/4 7-1a	B 3/4 7-1a

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. NPF-85

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 27, 2004, as supplemented by letters dated October 25, 2004, October 10, 2005, April 27, 2006, May 30, 2006, June 16, 2006, and August 4, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 146, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brooke D. Poole, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: August 23, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 146

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

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

Remove	Insert
1-2	1-2
1-6	1-6
1-7	1-7
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 3-16	3/4 3-16
3/4 3-31	3/4 3-31
3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
3/4 3-66	3/4 3-66
3/4 3-67	3/4 3-67
3/4 6-3	3/4 6-3
3/4 6-47	3/4 6-47
3/4 6-50	3/4 6-50
3/4 6-52	3/4 6-52
3/4 7-6	3/4 7-6
B 3/4 1-4	B 3/4 1-4
B 3/4 1-5	B 3/4 1-5
B 3/4 6-5	B 3/4 6-5
B 3/4 7-1	B 3/4 7-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 185 AND 146 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By application dated February 27, 2004 (Agencywide Documents and Management System (ADAMS) Accession No. ML040980153), as supplemented by letters dated October 25, 2004 (ML043000314), October 10, 2005 (ML053330355), April 27, 2006 (ML061230119), May 30, 2006 (ML061520487), June 16, 2006 (ML061780044), and August 4, 2006, Exelon Generation Company, LLC (Exelon or the licensee) requested changes to the Technical Specifications (TSs) for Limerick Generating Station, Units 1 and 2 (LGS).

The amendment revises the TSs to support application of an alternate source term (AST) methodology, with the exception that Technical Information Document (TID) 14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” will continue to be used as the radiation dose basis for equipment qualification.

2.0 REGULATORY EVALUATION

In December 1999, the U.S. Nuclear Regulatory Commission (NRC or the Commission) issued a new regulation, Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, “Accident Source Term,” which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their design-basis accident (DBA) analyses with an AST. Regulatory guidance for 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”. A licensee seeking to use an AST is required by 10 CFR 50.67 to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. The Exelon application of February 27, 2004, as supplemented, addresses these requirements by proposing to use the AST described in RG 1.183 as the design-basis source term in the evaluation of the radiological consequences of DBAs at LGS. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 as LGS's licensing basis for the DBA loss-of-coolant-accident (LOCA), the

control rod drop accident (CRDA), the fuel handling accident (FHA), and the main steamline break (MSLB).

This Safety Evaluation (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 on which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as further discussed in Regulatory Position 4.4 of RG 1.183. Except where the licensee has proposed a suitable alternative, the staff used the regulatory guidance in the following documents in performing this review:

- RG 1.183
- Standard Review Plan (SRP) Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”
- SRP Section 6.4, “Control Room Habitability Systems” (with regard to control room meteorology)
- SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Term”
- RG 1.23, “Onsite Meteorological Programs”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”
- RG 1.5, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors”
- Technical Specification Task Force (TSTF) Traveler TSTF-51, Revision 2 approved by the NRC on October 13, 1999, which provides for the relaxation of some TS requirements during refueling after a sufficient decay period has occurred.

The NRC staff also considered relevant information in the LGS Updated Final Safety Analysis Report (UFSAR) and TSs. Additionally, the NRC staff performed independent confirmatory calculations to evaluate the licensee’s revised design-basis dose analyses. The staff used the computer code RADTRAD, discussed in NUREG/CR-6604, “RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation,” and its supplements, in performing such independent dose calculations.

3.0 TECHNICAL EVALUATION

3.1 Accident Dose Calculations

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs performed by Exelon in support of this proposed license amendment. Exelon provided

information regarding these analyses in the February 27, 2004, submittal as supplemented by letters dated October 25, 2004, October 10, 2005, April 27, 2006, May 30, 2006, June 16, 2006, and August 4, 2006. The NRC staff held meetings with Exelon on July 14, 2005, and January 30, 2006.

The staff reviewed the assumptions, inputs, and methods used by Exelon to assess these impacts. The staff did independent calculations to confirm the conservatism of the licensee's analyses. However, the findings of this SE are based on the descriptions of the analyses and other supporting information submitted by Exelon. Only docketed information was relied upon in preparing this SE.

In accordance with the guidance in RG 1.183, a licensee is not required to re-analyze all DBAs for the purpose of the application, just those affected by the proposed changes. Exelon considered the following DBA events:

- LOCA
- FHA
- MSLB
- CRDA

The technical evaluation of these events is described below.

3.1.1 LOCA

The objective of analyzing the radiological consequences of a LOCA is to evaluate the design of various plant safety systems. These safety systems are intended to mitigate the postulated release of radioactive materials from the plant to the environment in the event that the emergency core cooling system (ECCS) is not effective in preventing core damage. 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. The primary coolant blows down through the 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 the 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.

3.1.1.1 Source Term

Exelon 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 (EFPDs) 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 EFPDs. The ORIGEN 2.1 computer code is acceptable to the NRC staff for estimating the core

inventory, as discussed in RG 1.183. The standard RADTRAD code default isotopes were used in the licensee's accident dose calculations for airborne radioactivity. To assure conservative shielding results, a total of 110 isotopes (including applicable isotopes from the 60 considered in the RADTRAD code) are used in the shine dose calculations.

Fission products from the damaged fuel are released into RCS and then into the primary containment. The release fractions and timing of these releases are given in Table 1.

3.1.1.2 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 reactor pressure vessel (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.

In lieu of modeling this transport mechanistically, Exelon has conservatively assumed that the fission product released from the fuel is homogeneously and instantaneously dispersed within the drywell free volume. For the duration of the 30-day accident, after the first 2 hours, the fission products are assumed to be homogeneously distributed between the drywell and the suppression chamber airspace. The NRC staff finds that the licensee's assumptions regarding drywell and containment mixing are consistent with assumptions previously found acceptable for a full implementation of an AST for the Perry Nuclear Power Plant, which has a Mark III containment, and has since been found acceptable for other boiling-water reactor (BWR) containments, most recently for Fermi 1. Exelon did not credit any reduction by suppression pool scrubbing for fission products transferred to the primary containment through the suppression pool.

Exelon 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. Exelon models this deposition using the 10^{th}-percentile values in the model described in the NRC staff-accepted NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (i.e., the "Powers Model"). Exelon 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 at 7.0 pH or higher. LGS proposes to use the standby liquid control (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 NRC staff reviewed the quantity of sodium pentaborate available with respect to the quantity of acid-producing debris and radiolytic acid production to confirm adequate pH control. In NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the NRC staff concluded that iodine entering the containment from the RCS during an accident would be composed of at least 95 percent CsI. Upon dissolution in the suppression pool, the predominant form of iodine would be the iodide ion (I-). The radiation-induced conversion of iodide in water into elemental iodine (I_2) is strongly dependent on the pH. Without pH control, a large fraction of the iodine dissolved in water in the ionic form will be converted to I_2 and released into the containment atmosphere. If the pH is maintained above 7, less than 1 percent of the dissolved iodine will be converted to I_2. Since the pH of the suppression pool is not controlled under normal conditions, I_2 may be released during a postulated LOCA when acids lower the pH.

To prevent the release of I_2 during a LOCA, an alkaline chemical capable of buffering the pH at a value above 7 must be added to the suppression pool. The LGS submittal proposes to do this by injecting sodium pentaborate from the standby liquid control system during a LOCA. The analysis assumes that the sodium pentaborate solution is injected and mixed in the suppression pool within 13 hours of the onset of a LOCA.

The licensee's calculations consider the effects of acids and bases created during a LOCA. The sump pH is affected by the generation of hydrochloric acid from the irradiation of electrical cables and nitric acid from the irradiation of water and air. Calculations for the amount of hydrochloric and nitric acids generated are based on guidelines in NUREG-5950, "Iodine Evolution and pH Control." The model for the cables in containment maximizes the generation of hydrochloric acid in order to provide a conservative value.

The staff reviewed the licensee's methodology, assumptions, and calculations for determining the 30-day post-accident pH value and found the evaluation acceptable. The NRC staff independently verified through calculations that the sump pH is maintained above 7 for the duration of the 30-day period.

The staff reviewed the SLC system with respect to the newly proposed SLC system role in delivery of sodium pentaborate to the suppression pool for pH control. 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.

The Containment and Ventilation Branch (SCVB) reviewed the SLC system with respect to the SLC role in delivery of sodium pentaborate to the suppression pool for pH control. 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. The licensee stated that the SLC is designated a safety-related system.

The staff reviewed the licensee's submittals and the responses to requests for additional information (RAIs) on the use of the SLC system for the safety-related function. From the licensee's statements, the NRC staff has concluded the following:

> The SLC system is designated a safety-related system. As such, the SLC system as designed and installed, is a high quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation, specifically:

1. The system components required for reactivity control and new suppression pool pH control functions are seismically qualified.

2. The system is provided with emergency power with the capability to supply power from the emergency diesel generators.

3. The system is subject to American Society of Mechanical Engineers (ASME) Section XI, inservice inspection requirements as required by 10 CFR 50.55a, “Codes and Standard.”

4. The system is within the scope of 10 CFR 50.65, the Maintenance Rule.

5. Most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses. The exceptions are the containment isolation check valves. This is discussed below under single failure review.

6. Emergency Operating Procedures (EOPs) direct the activation of the SLC following a LOCA when the reactor water level cannot be maintained above the top of active fuel. Manual initiation of SLC is also directed in the severe accident management guidelines (SAMGs), which are entered when adequate core cooling cannot be maintained. Procedures will be updated to specifically direct boron injection without dilution until the required amount of boron is injected for pH control.

7. Training will be provided on the new SLC injection function as part of operator re-qualification training and EOP and SAMG training.

The NRC staff considered components that could be subject to single failure. The licensee identified two components, the containment isolation check valves on the injection line, as not being redundant. The containment isolation valves are Borg-Warner Lift Check Valves Model 77930 mounted in the injection line. In the periodic inspections and testing of these valves, LGS has not experienced any failures of these valves to open on demand. A review of the industry databases, EPIX and NPRDS, was performed and no failures of check valves of this type and manufacture to open were identified. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the staff has determined that the potential for failure of this component is very low based on the quality as established by its procurement, periodic testing and inspection, and its historical performance. Therefore, the staff finds that the use of a single penetration of the containment with the identified check valves as described by the licensee to be acceptable.

The staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the reactor vessel. The transport of reactor vessel contents including the sodium pentaborate to the suppression pool is by flow through the break (assumed to be a large recirculation pipe break) to the drains that feed the suppression pool. Core Spray systems and low pressure coolant injection (LPCI) systems are used to maintain water level and assure core cooling after a LOCA. Procedure changes will be implemented to assure that the SLC system is initiated when there is the indication of fuel damage. Procedure changes will also be made to modify the suppression pool cooling return flow path. The changes will assure that LPCI flow is maintained through the

vessel in order to assure that the sodium pentaborate is swept from the reactor vessel to the suppression pool.

Using the LPCI in this mode for suppression pool cooling also provides mixing. The NRC staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution.

The specific changes being made to TS Section 3/4 1.5 are the extension of applicability to Operational Condition 3 with one pump and the associated flow path available, and adding a new action to restore the system to operable status in 8 hours or be in HOT SHUTDOWN in the next 12 hours and be in COLD SHUTDOWN in the next 24 hours. The extension of applicability to Operational Condition 3 provides the capability of injecting SLC during hot standby. This would not be necessary for the anticipated transients without scram (ATWS) function of the SLC but is reasonable for the LOCA pH control function. Clarifying the action and response time is appropriate for this action. On the basis of the above discussion, the staff finds these changes acceptable.

The licensee also proposed changing Surveillance Requirement (SR) 4.1.5.b.2 to reflect 185 lbs of Boron-10 in lieu of 3754 lbs of sodium pentaborate. The NRC staff's review of the licensee's submittal showed that the licensee calculated that 240 lbs of total boron is the minimum amount required to maintain post-LOCA sump pH above 7. The TS change would require a minimum of 185 lbs of the Boron-10 isotope. The equation listed under TS 4.1.5.b.2 ensures that the requisite enrichment of Boron-10 is 29%. At this enrichment of Boron-10, and a minimum Boron-10 weight of 185 lbs, the total amount of boron in the system is greater than the 240 lbs required to maintain a pH of 7 in LGS's post-LOCA sump environment. For the Boron-10 enrichment required, the TS requirement for 185 lbs of Boron-10 will ensure that the sump pH is maintained above 7 for 30 days following a LOCA, and will prevent the release of radioactive iodine. As such, this change to the SR is acceptable.

3.1.1.3 LOCA Release Pathways

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

- Design leakage of primary containment atmosphere.
- Design leakage through main steamline isolation valves (MSIVs).
- Design leakage from ECCS piping and components that recirculate suppression pool water outside of the primary containment.

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, Exelon 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.

3.1.1.4 Containment Leakage Pathway

The drywell and wetwell are projected to leak at their design leakage of 0.5 percent of their atmospheric contents by weight per day for the first 24 hours, and 0.25 percent of their atmospheric contents by weight for the remainder of the 30-day accident duration. Leakage from the drywell and wetwell will collect in the free volume of the secondary containment and be released to the environment via ventilation system exhaust or leakage. Following a LOCA, the standby gas treatment system (SGTS) fans start and drawdown the secondary containment to create a negative pressure with reference to the environment beginning at 13.5 minutes after the onset of the gap release (15.5 minutes after the onset of the LOCA). After 15.5 minutes, credit for filtration through the reactor enclosure recirculation system (RERS) and the SGTS is taken. All releases from the reactor enclosure to the environment are modeled as ground-level releases.

The leakage from the primary containment is released into the secondary containment (i.e., reactor enclosure). RG 1.183, Appendix A, Position 4.2, states that primary containment leakage is assumed to be released directly to the environment during any time period in which the secondary containment does not have a negative pressure, as defined in the TSs. The licensee proposed to take an exception to this guidance by taking credit for limited mixing in the secondary containment due to the operation of the RERS. RG 1.183, Appendix A, Position 4.4, states that credit for dilution in the secondary containment may be taken when adequate means to cause mixing can be demonstrated. This dilution credit for mixing should generally be limited to 50 percent. The staff interprets this to mean that the leakage from the primary containment is released into a volume that is 50 percent of the free volume of the secondary containment.

The licensee's analysis takes credit for limited mixing within the reactor enclosure by the engineered safety feature (ESF) RERS after initiation of the system at 3 minutes post-LOCA, but before the secondary containment is drawn down to negative pressure to the outside environment at 15.5 minutes.

The licensee's evaluation of mixing accounts for the fact that the primary containment releases to the reactor enclosure are continuous, but vary in activity. Exelon's analysis assumed that the mixing into the secondary containment was capped at 50 percent, in accordance with guidance in RG 1.183. In response to the NRC staff's RAIs, the licensee's discussion of the modeling of mixing in the reactor enclosure by the RERS during secondary containment drawdown was provided by letter dated August 4, 2006.

The licensee requested that credit for mixing by the RERS be allowed during the 15.5 minute drawdown of the secondary containment. The credit for mixing would be used in design-basis analyses to reduce the primary containment released to the environment and, thus, reduce the overall dose results of the calculation.

The staff has reviewed this request. SRP 6.5.3 and RG 1.183 specifically state that credit should not be taken for mixing in the secondary containment until the drawdown period is complete, and that releases from the primary containment to the secondary containment during drawdown should be treated as a release to the environment.

The licensee requested a non-standard review on the basis that the LGS facility is unique in that it has a RERS system which recirculates the air in the secondary containment at

60,000 cfm and thus promotes mixing. The licensee proposed, in its original submittal, to use a credit of 50 percent mixing (the source term diluted by 50 percent of the volume of the secondary containment). After discussions with the staff, the licensee proposed, by letter dated June 16, 2006, to use a step-wise increase of mixing from 0 to 50 percent over 22.5 minutes and to freeze the value at 50 percent for the remainder of the release period. The licensee submitted a calculation as Attachment 3 to the referenced letter to establish the basis for its mixing efficiencies per time step.

After careful review of the information presented, the NRC staff concluded that the calculation did not provide a sufficient technical basis. No source for the formula in the calculation was presented. The calculation appeared to estimate mixing on the basis of an air exchange ratio of the 60,000 cfm RERS operating in a 1,800,000 cubic foot volume. The calculation did not consider the continuous input of leakage from the primary containment to the secondary containment. At best, the calculation represents the mixing efficiencies of a single puff of contaminants as it is mixed with the secondary containment atmosphere over time.

The staff considered the uniqueness of the LGS configuration with respect to the RERS. The RERS, with 125 return registers and 150 supply registers located throughout the secondary containment volume, would promote substantial mixing of the secondary containment atmosphere. Although secondary containment is not established until the completion of drawdown, the secondary containment does define a volume in which some degree of mixing could occur.

In response to further discussions with the NRC staff during a teleconference on July 18, 2006, the licensee provided additional information in a supplement dated August 4, 2006. The letter states that although the requested mixing efficiencies are based on a single release, the continuous release is accounted for by evaluating the time of residence of each interval of release at the mixing efficiency of that time interval, using RADTRAD to adjust for the non-linear effects of the release, and summing the results to get an overall dose. In addition, the licensee stated that a single effective mixing for the drawdown period as determined from a review of the results would be approximately 7 percent.

The NRC staff concurs that an overall mixing efficiency of 7 percent is reasonable, considering the operation and capabilities of the RERS. In addition, the NRC staff accepts the methodology of accounting for continuous release by a step-wise summation of the release in each interval evaluated at the mixing efficiencies for that interval. The staff also accepts the mixing efficiencies for the interval based on the formula presented in the June 16, 2006, letter averaged over the interval as used in the licensee's methodology. Based on the uniqueness of the LGS configuration, which includes a RERS system with its substantial mixing capability, the staff finds that a limited amount of mixing credit during drawdown not to exceed a calculated effective mixing percentage of 7 percent as determined by the method established in the August 4, 2006, letter is acceptable for RERS system operation.

The NRC staff also reviewed the requirement for an additional technical specification based on 10 CFR 50.36(c)(2)(ii)(c) criterion 3. Since the RERS system becomes a system that is on the success path for mitigating a DBA (because of the mixing credit), a limiting condition for operation (LCO) is required. The existing SR 4.6.5.4.b.3, which tests the subsystem flow rate and demonstrates that the RERS does recirculate at 60,000 cfm, satisfies this requirement since the degree of mixing is dependent on RERS flow.

3.1.1.5 Secondary Containment Bypass Leakage

A source of containment leakage that bypasses the secondary containment is MSIV leakage. The model for this leakage is discussed below.

MSIV leakage limits of 100 standard cubic feet per hour (scfh) per valve, or 200 scfh for four steamlines, at a test pressure of 36.7 psia (22 pounds per square inch gauge) are contained in the LGS TSs. Since the allowable TS leakage is assessed in units of scfh, and the steamlines are not at standard temperature and pressure conditions, Exelon adjusted the assigned flow rates for the assumed accident conditions.

In a May 30, 2006, supplement, Exelon responded to NRC staff Question 4a regarding the adjustment to the MSIV leakage for a main steamline segment upstream of the inboard MSIV. The staff asked Exelon why the temperature used to adjust the flow rate in this volume did not include the effects of the temperature on the steamline and the reactor vessel. The accident condition flow rates are not well understood, and for this reason the NRC staff is reluctant to accept using temperatures less than the steamline temperature to correct the assumed flow rate. In response to NRC staff Question 4a, Exelon performed a sensitivity study of the impact of their modeling of the MSIV flow, which did not use the steamline temperature. Based upon this sensitivity study, the NRC staff has reasonable assurance that for the LOCA analysis, as modeled, not using the steamline temperature does not affect the conclusions of the Exelon analysis.

Exelon assumes a maximum MSIV leakage of 100 scfh in the broken line. The unbroken line is assumed to leak at 100 scfh, and the other two lines are assumed not to leak. These leak rates are consistent with the LGS TSs, which limit MSIV leakage to 200 scfh for all steamlines and 100 scfh for any one line. The leak rates are assumed to decrease to 55.1 percent after 24 hours.

Exelon assumes that the outboard MSIVs fail to close on all four main steamlines with one line broken upstream of the inboard MSIV. In an October 10, 2005, supplement, Exelon responded to NRC staff Question 11e, confirming that these failures represented the limiting MSIV failures.

Exelon assumes three aerosol settling volumes (nodes) for the unbroken main steamline and two settling nodes for the broken pipe. The main steamline assumed to be broken does not have the volume between the RPV and the inboard MSIV available for iodine removal, so it only assumes two aerosol settling nodes.

Exelon'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. 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." 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.

Exelon's modeling of aerosol settling in the MSIV leakage pathway for LGS is different from that for Perry in that LGS credits five volumes (one pathway used two volumes in the steamline plus the condenser, the other pathway used one volume plus the condenser) for determining the settling deposition[1] while, in contrast, Perry credited three volumes.[2] The NRC staff questioned whether assuming the same settling velocities throughout the piping system and the condenser was adequately conservative. The staff's concern was that the removal through aerosol settling was overestimated by modeling all volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping. This lesser settling effect is due to larger and heavier aerosols that settle out of the main steamline atmosphere in upstream sections of piping.

Exelon responded by changing its model to combine penetration piping and downstream piping into a single outboard node. Additionally, Exelon used a 20-group probability distribution of settling velocities with efficiencies determined for each group and a net weighted average efficiency (a process that Exelon states is significantly more conservative than use of a median settling velocity). Exelon 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, however, because of recent concerns regarding the AEB-98-03 report and the lack of further information, does not know how much deposition (i.e., which settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, but included some additional conservatism to attempt to address the NRC staff's questions on the applicability of the AEB-98-03 methodology to LGS. The NRC staff has performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and has found that the total LOCA dose from all main steamline pathways would continue to be acceptable, even when the AEB-98-03 10th-percentile settling velocity is assumed for bypass aerosol settling. 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 use of the median settling velocity noted as reasonable in AEB-98-03. Given this information, and the presence of a seismically-qualified condenser, the NRC staff finds the LGS main steamline aerosol settling model to be reasonable.

Exelon also assumed deposition of I_2 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," which the NRC staff has found acceptable, as discussed in RG 1.183. The Cline Report provides I_2 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, Exelon assumed I_2 deposition occurs on the entire surface area of the horizontal and vertical piping. Exelon 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 release instantly. No removal of organic iodine was assumed by the licensee.

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1 This is treated as an effective filtration efficiency in both models.

2 No credit is taken for deposition in the MSIV alternate drain pathway to the condenser.

3.1.1.6 Secondary Containment Filtration Systems Modeling

The licensee's dose analysis transport modeling included dilution into the free volume of the reactor enclosure building. As discussed above, the RERS provides for mixing of the air in the reactor building after initiation. The RERS initiates at 3.5 minutes. After 15.5 minutes, the secondary containment is drawn down to subatmospheric pressure, at which time the licensee's analysis assumes filtration by the RERS until the end of the accident, with filter efficiencies of 99 percent for particulates and 95 percent for elemental and organic iodine.

The SGTS was assumed to have a flow rate of 3000 cubic feet per minute (cfm) until the secondary containment is drawn down at 15.5 minutes. After that time, the SGTS flow rate is 2500 cfm until the end of the accident. The SGTS filter efficiency for all forms of iodine and for particulates is 99 percent after drawdown at 15.5 minutes, while no filtration is assumed before drawdown.

3.1.1.7 ESF Leakage

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. For the purposes of assessing the consequences of leakage from the ESF systems, Exelon assumed that all of the radioiodines released from the fuel are instantaneously moved to the suppression pool. This source term assumption is conservative, in that all of the radioiodine released from the fuel is available for both primary containment atmosphere leakage and the ESF system leakage. Exelon assumes that 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release as 97-percent elemental and 3-percent organic iodine. The release continues for 30 days. The NRC staff finds these assumptions to be consistent with the guidance of RG 1.183 and, therefore, acceptable.

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. Consistent with RG 1.183, this 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 at 0 minutes after the event.

The second source of potential ESF leakage is into the condensate storage tanks. Exelon performed an analysis which concluded that the dose from this pathway is negligible.

3.1.1.8 Control Room

Exelon evaluated the dose to the operators in the control room. While in the radiation isolation mode of operation, the control room emergency fresh air supply system (CREFAS) is designed to maintain the control room at a positive pressure of at least 1/8 inch water gauge relative to its surrounding areas with a filtered outside air flow rate of less than, or equal to, 525 cfm. Verification and testing to determined the control room unfiltered inleakage with the CREFAS system operation in the radiation isolation mode was performed, as required by Generic Letter 2003-01, "Control Room Habitability." Tracer gas test results determined that the filtered outside air flow rate was less than 525 cfm and the unfiltered inleakage into the control room

was less than 100 cfm for both trains of CREFAS. The tracer gas test results were formally submitted to the NRC in a letter dated December 10, 2004 (ML043510210).

In an RAI dated August 18, 2005, the NRC staff asked Exelon if its dose analysis factored the impact of ingress and egress into the total assumed unfiltered inleakage to the control room. Exelon's response, dated October 10, 2005, to NRC Question 21, stated that the unfiltered inleakage value assumes that there will be 0 cfm for ingress and egress. This is based on installation of a main control room door seal, as discussed in LGS UFSAR Sections 1.13, 6.4, and 15.10. In Exelon's response, dated May 30, 2006, to NRC Question 9c, Exelon provided further clarification regarding the unfiltered inleakage assumed for ingress and egress. Exelon stated that during the tracer gas test to determine the control room inleakage rate, the door seal was not installed and access to the control room was not restricted. The maximum measured inleakage rate of 77 cfm includes the potential inleakage from opening of the control room doors (allowable inleakage rate of 275 cfm). The 10 cfm unfiltered inleakage due to ingress/egress as recommended in SRP 6.4 is included in the 225 cfm design analysis value (note: this was changed from 275 to 225 in the June 16, 2006, supplement). The acceptance criteria for tracer gas testing will, therefore, be 215 cfm to reflect the assumption of 10 cfm for ingress/egress. Based upon using an assumed value of 10 cfm for ingress/egress, the NRC staff finds this assumption acceptable for all accidents analyzed.

Control room intake and recirculation filtration by the CREFAS are credited in the licensee's LOCA dose analysis with automatic initiation of the radiation isolation mode on a high radiation signal in the control room intake. The analysis used intake and recirculation filter efficiencies of 99 percent for aerosols and 95 percent for elemental and organic iodine. The bounding total flow through the CREFAS filters is 3000 cfm - 10 percent = 2700 cfm. The CREFAS has two operating modes. In the radiation isolation mode, 525 cfm of the CREFAS flow is filtered outside air intake, with the remaining 2175 cfm being filtered recirculated air from the control room. In the chlorine isolation mode, the entire 2700 cfm is recirculated air. Exelon also performed calculations modeling the control room in chlorine isolation mode and found that the dose in the control room with the CREFAS in radiation isolation mode is bounding.

3.1.1.9 Direct Gamma Shine Doses

For LGS, the post-LOCA direct shine dose from sources outside of the control room is dominated by the gamma shine from an LGS Unit 1 core spray pipe with a 14-inch normal pipe size, located 18 inches from the 36-inch thick shield wall between the control room and the reactor enclosure.

The dose from the pipe has been re-evaluated for AST-based emergency core cooling system fluid radionuclide concentrations. The dose was integrated over the accident duration, with RG 1.183 control room occupancy factors for the 1-to-4 day and 4-to-30 day periods. A total of 110 isotopes were evaluated in determining doses from shielded ECCS piping to control room areas accessible to personnel. The resulting maximum integrated dose from this pipe is 1.48 roentgen equivalent man (rem) at 1 foot from the interior surface of the control room perpendicular to that surface.

Other external sources were also evaluated. The only other major dose contribution was for a residual heat removal pipe located 50 to 60 feet from the control room. The calculated dose inside the control room from this pipe is 0.18 rem. The shine dose from reactor enclosure

airborne activity was also evaluated, and determined to be 0.039 rem. The reactor enclosure recirculation system filters, SGTS and CREFAS filters dose shine contributions were determined to be negligible by Exelon.

3.1.1.10 LOCA Summary

The 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 NRC staff are presented in Table 1. Based upon the information provided by Exelon, the staff finds that the licensee used analysis, methods, and assumptions consistent with the guidance of RG 1.183, except as otherwise discussed and accepted above. The 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 staff finds, with reasonable assurance, that the licensee's estimates of the exclusion area boundary (EAB), low-population zone (LPZ), and control room doses for the LOCA will continue to comply with these criteria.

3.1.2 FHA

The FHA analysis postulates that a spent fuel assembly is dropped during refueling operations. Two cases were evaluated. These cases involved a drop of a fuel assembly in the reactor well or over the spent fuel pool.

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 32.3 feet for the fuel assembly and 47 feet 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. In accordance with the current licensing basis, the limiting design-basis analysis conservatively postulates to damage 212 pins (based upon a fuel assembly with an 8 x 8 fuel pin array and a peaking factor (PF) of 1.7[3]). A post-shutdown 24-hour decay period was used to determine the release activity inventory.

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 two hours, without credit for reactor building filtration or dilution. The control room was modeled without taking credit for automatic system actuation. Although the normal maximum flow into

[3] Other fuel types were considered and the 8 x 8 assembly with a peaking factor of 1.7 was found to be limiting. The other fuel types include a 7 x 7 fuel with a 1.5 PF, GE11 and GE13 (9 x 9 fuel) with a PF of 1.5 and GE 12 and GE14 (10 x 10 fuel) with a PF of 1.7.

the control room is 2100 cfm, a control room changeover rate of 1 control room volume (126,000 cfm) is used to allow for any unfiltered inleakage (including 10 cfm for ingress and egress).

The exhaust point under the assumed condition of no filtration is the reactor building south stack. The north stack, which is used for releases filtered by the SGTS, is located closer to the control room intake and, therefore, has a higher atmospheric dispersion factor. The SGTS is designed to remove 99 percent of the iodine that would otherwise be released; this filtration more than overcomes the effect of the higher atmospheric dispersion. Therefore, using the south stack unfiltered release in the dose analysis is bounding for a release through the SGTS and north stack. Crediting the SGTS for this analysis requires the SGTS to remain operable during fuel handling. To assure that any release through the north stack remains filtered for an FHA, LGS TS 3.6.5.3 denotes SGTS operability requirements (as stated in the May 30, 2006, supplement, Question 10 and Attachment 2) for movement of irradiated fuel, core alterations, or during operations with the potential to drain the vessel.

Site walkdowns and specific review of the LGS general arrangement drawings such as M-102, the plan at Elevation 217 feet and 0 inches (one foot above grade), and M-107, the section showing the north and south stack, confirmed that there are no potential release pathways that could be worse with respect to the control room intake than the analyzed stacks. In particular, there are no hatches or single-door reactor building openings leading directly to the outside, and any grade openings are considered to have atmospheric dispersion factors that are bounded by the south stack release point atmospheric dispersion factor, based on the greater distances of travel required for releases from them to the control room intake. This evaluation includes the large railroad doors at grade elevation, which could be postulated to be open at the same time as the equipment hatch cover on the refueling floor to support a future spent fuel cask move.

3.1.2.1 Fuel Transfer Area or Spent Fuel Pool Drop

Exelon evaluated a second case involving a drop over the spent fuel pool. Exelon stated that the postulated activity released would be lower than the limiting case described above based on the following.

The drop height over the spent fuel racks is approximately 1.2 feet. At a drop height of 1.2 feet, the kinetic energy available to cause fuel damage is reduced. The number of pins damaged in the design-basis drop over the reactor vessel (fuel well) would bound the number of pins damaged in a drop elsewhere because the drop height is significantly greater in the drop over the fuel well.

Exelon performed an assessment of the number of pins which would fail due to a postulated drop over the spent fuel pool. Exelon used a drop height of 2.33 feet in this analysis. Exelon estimates that 76 pins (based upon an 8 x 8 assembly) would fail by this scenario. The TS minimum required water depth available over an assembly lying on the fuel assembly bails is approximately 21.6 feet for a dropped bundle over the spent fuel pool. Exelon determined the difference in decontamination factor (DF) for the reduced coverage of water. Based on the DF and the number of pins postulated to fail, Exelon concluded that the consequences of an FHA over the reactor well bound those for an FHA over the spent fuel pool.

3.1.2.3 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 1. Based upon the information provided by Exelon, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183. The 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 staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses for the FHA will continue to comply with these criteria.

3.1.3 MSLB

The postulated MSLB accident assumes a double-ended break of one main steamline outside the primary containment with displacement of the pipe ends that permit maximum blowdown rates. Two activity release cases corresponding to the pre-accident elevated iodine concentration (spike) and maximum equilibrium concentration allowed by TSs of 4.0 µCi/gm and 0.2 µCi/gm dose equivalent iodine -131, respectively.

The mass of coolant released is the amount in the steamline and connecting lines at the time of the break, plus the amount passing through the MSIVs prior to closure (5.5 seconds[4]). A total of 140,000 pounds per minute (lbm) of liquid is assumed to be released during blowdown.

The analysis assumes an instantaneous ground level release. No holdup or dilution by mixing in the turbine enclosure air volume is credited. The released reactor coolant is assumed to expand to a hemispheric volume of 42,514 cubic meters at atmospheric pressure and temperature (consistent with an assumption of no turbine enclosure holdup credit). The activity in the release hemispheric volume is assumed to be equal to the activity present in the 140,000 lbm of liquid released. Noble gas releases correspond to a 0.35 Ci/sec offgas release rate after a 30-minute delay.

Exelon assumed that the iodine species released from the main steamline are 95 percent CsI as an aerosol, 4.85 percent elemental, and 0.15 percent organic as stated in RG 1.183, Appendix D, Regulatory Position 4.4. Consistent with this position, Exelon included cesium activity, as CsI, in the release. Exelon assumed that one atom of cesium accompanies each of the iodine atoms for 95 percent of the total iodine release. The licensee determined the cesium isotopic abundance (ratio of each isotope to the total cesium release) for longer lived or stable isotopes, based on source terms developed for pH control at LGS. The isotopic abundance for shorter lived isotopes was based on the source term in the American National Standards Institute/American Nuclear Society Standard ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors."

No credit is taken for the operation of the CREFAS. Inhalation doses are determined based on concentrations at the control room intake, and exposures for the duration of the hemispheric

[4] In an Exelon response, dated October 25, 2004, to NRC Question 3, the licensee stated that its original request to increase the MSIV closure time was to be deleted from the submittal. Therefore, the current licensing basis value of 5.5 seconds for MSIV closure is retained for the purpose of the radiological analysis.

volume traversing the intake. External exposure doses are determined based upon concentrations at the intake during the duration of the hemispheric volume traversing the intake. The inhalation committed effective dose equivalent (CEDE) dose conversion factors are from Federal Guidance Report 11 and the external effective dose equivalent dose conversion factors are from Federal Guidance Report 12.

3.1.3.1 MSLB 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 1. Based upon the information provided by Exelon, the NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183. 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, that the licensee's estimates of the EAB, LPZ, and control room doses for the MSLB will continue to comply with these criteria.

3.1.4 CRDA

The design-basis CRDA assumes the rapid removal of the highest worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. For the dose consequence analysis, it was assumed that 1200 of the fuel rods in the core were damaged, with melting occurring in 0.77 percent of the damaged rods. A core average radial peaking factor of 1.7 was used in the analysis. For releases from the breached fuel, 10 percent of the core inventory of noble gases and iodines are assumed to be in the fuel gap. For releases attributed to fuel melting, 100 percent of the noble gases and 50 percent of the iodines are assumed to be released to the reactor coolant.

Instantaneous mixing of the activity released from the fuel in the reactor coolant is assumed, with 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the remaining radionuclides that are released into the reactor coolant, to reach the turbine and condenser. Of this activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodines, and 1 percent of the particulate radionuclides are available for release to the environment. The main condenser is assumed to leak activity into the turbine enclosure at a rate of 1 percent of the condenser volume per day. This leaked activity is then released, unfiltered, to the environment by way of the north stack, taking no credit for holdup in the turbine enclosure. The forced flow path through the steam jet air ejectors discharges to the off-gas system. This pathway credits elimination of iodine releases and delay of noble gas releases by the off-gas system charcoal delay beds. This credit is as currently used and licensed in conformance with NEDO-31400A. Unfiltered release from the turbine enclosure is via the north stack at the rate of 1.0 percent of the condenser activity per day for 24 hours.

The LGS Units 1 and 2 control room is modeled as a closed volume of 126,000 cubic feet. The control room was modeled without taking credit for automatic system actuation. Although the normal maximum flow into the control room is 2100 cfm, a control room changeover rate of one control room volume (126,000 cfm) is used to allow for any unfiltered inleakage (including 10 cfm for ingress and egress). No credit is taken for any filtration of intake flow into the control room.

3.1.4.1 CRDA 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 1. Based upon the information provided by Exelon, the staff finds that the licensee used analysis methods and assumptions are consistent with the guidance of RG 1.183. The 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 staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses for the CRDA will continue to comply with these criteria.

3.2 Atmospheric Dispersion Estimates

The licensee calculated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of DBAs on the control room and offsite exposures. The licensee used the ARCON96 and PAVAN atmospheric dispersion computer models to calculate χ/Q values for the LOCA, the CRDA, and the FHA. In addition, the licensee used a default set of meteorological conditions to calculate χ/Q values for the MSLB accident. The resulting set of control room and offsite (EAB and LPZ) χ/Q values represents a change from the values currently presented in the LGS UFSAR.

3.2.1 Meteorological Data

The licensee generated the new LOCA, CRDA, and FHA control room and offsite χ/Q values for this amendment application, using meteorological data collected at the LGS site during the period 1996–2000. The licensee provided these data in the form of hourly data files (for input into the ARCON96 computer code) and a joint wind speed, wind direction, and atmospheric stability frequency distribution (for input into the PAVAN computer code) in Attachment 7 to its application, dated February 27, 2004.

The licensee stated that its onsite meteorological measurement program meets the guidelines of RG 1.23. The licensee performed its atmospheric dispersion analyses using wind measurements taken at 9.1 meters (30 feet) and 53.3 meters (175 feet) above ground level (AGL) on the main meteorological tower (Tower 1). The licensee determined atmospheric stability using the temperature difference measurements between the 52.1-meter (171-foot) and 7.9 meter (26-foot) levels on the main meteorological tower. The licensee selected this 1996–2000 data set from 31 years of available historic onsite data records (1972–2002) because this period constituted a high data recovery rate for a data set that the licensee considers to be representative of the site. To ensure a high data recovery rate, the licensee substituted wind speed and direction data from the backup meteorological tower (Tower 2) for missing wind data from the main meteorological tower. The resulting combined data recovery rate of wind speed, wind direction, and stability data exceeded the RG 1.23 goal of 90 percent for each of the 5 years (1996–2000).

The NRC staff performed a quality review of the 1996–2000 hourly meteorological database provided by the licensee 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. Wind speed, wind direction, and stability class frequency distributions were reasonably similar year to year and the 1996–2000 wind

direction and stability class frequency distributions were reasonably consistent with the 1972–1976 data presented in Section 2.3.2 of the LGS UFSAR. The time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). A comparison of joint frequency distributions derived by the NRC staff from the ARCON96 hourly data with the joint frequency distributions developed by the licensee for input into the PAVAN atmospheric dispersion model showed reasonable agreement.

For the reasons cited above, the NRC staff concludes that the 1996–2000 LGS meteorological database provides an acceptable basis for making atmospheric dispersion estimates for use in the DBA dose assessments performed in support of this application for amendment.

3.2.2 Control Room Atmospheric Dispersion Factors

LGS has a common control room facility for both units which is served by one control room air intake. The licensee modeled all activity entering the control room (including unfiltered inleakage) as though it came through the control room air intake. That is, the control room air intake χ/Q values were considered to be bounding for all potential inleakage pathways to the control room.

The licensee assumed LOCA and CRDA releases were discharged to the environment through the north stack and the FHA releases were discharged to the environment through the south stack.[5] The licensee calculated control room air intake χ/Q values for the LOCA, CRDA, and FHA events using the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses.

The licensee executed ARCON96 using 1996–2000 hourly 9.1-meter and 53.3-meter wind speed and direction data, and hourly stability class data from the main onsite meteorological

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5 The staff asked the licensee in Question 23 of the RAI letter dated August 18, 2005, to justify that an FHA release through the south stack is an appropriately conservative assumption, given that the secondary containment may be inoperable during the movement of fuel assemblies that have a decay period of at least 24 hours. In its response, dated October 10, 2005, the licensee stated that the north stack is closer to the control room air intake and, therefore, has higher χ/Q values. However, the north stack is only used for releases filtered by the SGTS, which is designed to remove at least 99 percent of the iodine that would otherwise be released. The licensee stated that this filtration more than overcomes the effect of the higher north stack χ/Q values and, therefore, the south stack unfiltered release is bounding. The licensee also stated that, other than the north stack, there are no potential release pathways that could be worse than the south stack, with respect to the control room intake, because such openings (e.g., hatches and reactor enclosure openings leading directly to the outside and large railroad doors at grade elevation) are further from the control room air intake than the south stack release point. The staff finds this response acceptable.

The NRC staff asked the licensee in Question 10 of the RAI letter, dated April 27,2006, to justify how the FHA dose calculation can credit the SGTS filtration, given that the licensee has requested to remove the TS 3.6.5.3 SGTS operability requirements during refueling. In its response, dated May 30, 2006, the licensee agreed to revise the SGTS TS 3.6.5.3 to restore the SGTS operability requirements when moving irradiated fuel, performing core alterations, or during operations with the potential to drain the vessel.

tower. The release heights were set equal to the height of the north and south stacks (61 meters AGL) and the intake height was set equal to the height of the control room air intake (37.8 meters AGL). Because the release heights of both stacks are less than 2½ times the height of adjacent buildings, RG 1.194 states that both release locations should be modeled using the ARCON96 ground level release option. The licensee effectively modeled both release locations as ground level releases by setting the stack exit velocities to zero. The resulting control room χ/Q values for the LOCA, CRDA, and FHA events are presented in Table 2.

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 the ARCON96 model for the LGS site. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and site practice, except that not all the ARCON96 default data values specified in RG 1.194 were used by the licensee in its analysis (i.e., the licensee used surface roughness length, minimum wind speed, and averaging sector width constant values of 0.10 meters (m), 0.22 meters per second (m/s), and 4.0, respectively, in lieu of RG 1.194 specified values of 0.20 m, 0.5 m/s, and 4.3, respectively). Nonetheless, the NRC staff performed a comparison evaluation which supported the acceptability of the licensee's ARCON96 χ/Q values.

The licensee modeled atmospheric dispersion for the MSLB event assuming a double-ended break of one main steamline outside the primary containment with displacement of the pipe ends that permits maximum blowdown rates. The release from the break to the environment was assumed to be instantaneous. The licensee assumed the resulting steam cloud consisted solely of the initial steam blowdown and of the portion of the liquid reactor coolant release that flashed to steam. The resulting released reactor coolant and steam were assumed to expand to atmospheric pressure, forming a hemispheric cloud that was assumed to move downwind at a speed of 1 meter per second. The licensee took no credit for buoyant rise of the steam cloud and no dilution of the steam cloud by ambient air entrainment was assumed to occur as the cloud moved downwind. The licensee assumed the activity in the steam cloud entered the control room via the control room air intake while the cloud remained over the control room air intake. Control room χ/Q values do not apply to the MSLB control room dose assessment in that the licensee developed equivalent χ/Q values in the spreadsheet it used to perform the MSLB control room dose calculation.

The NRC staff has reviewed the licensee's assessments of control room post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting control room χ/Q values are presented in Table 2. On the basis of this review, the staff concludes that the χ/Q values for DBA releases to the LGS control room, as presented in Table 2, are acceptable for use in the DBA CRDAs performed in support of this amendment application.

3.2.3 <u>Offsite Atmospheric Dispersion Factors</u>

The licensee used the PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") to calculate χ/Q values for the LOCA, CRDA, and FHA events at

downwind distances of 731 meters for the EAB and 2043 meters for the LPZ. The PAVAN computer code implements the guidance provided in RG 1.145.

As discussed previously, the licensee assumed the LOCA, CRDA, and FHA releases were discharged to the environment through either the north stack or the south stack. Because the height of both release locations is less than 2½ times the height of adjacent buildings, the licensee modeled the stack releases as ground level releases, in accordance with RG 1.145 guidance. The licensee modeled building wake effects based on a reactor building vertical cross-sectional area of 2426 square meters.

The licensee's meteorological input to PAVAN consisted of a joint frequency distribution of wind speed, wind direction, and atmospheric stability data for the period 1996–2000. Wind speed and direction data from the main meteorological tower's 9.1-meter level were used. Stability class was based on the temperature difference data between the 52.1-meter and 7.9-meter levels on the main meteorological tower.

The NRC staff evaluated the applicability of the PAVAN model and concluded that there are no unusual release characteristics, plant configuration, or site topography that preclude the use of the PAVAN model for the LGS site. The NRC staff qualitatively reviewed the inputs to the PAVAN computer runs and found them generally consistent with site configuration drawings and staff practice. PAVAN produces the best results if the wind speed data are classified into a large number of categories at the lower wind speeds. Consequently, the staff also reran the PAVAN code using a joint frequency distribution with a larger number of low wind speed categories (generated by the staff from the licensee's hourly ARCON96 meteorological database) and determined that the licensee's PAVAN computer runs produced conservative results. The resulting EAB and LPZ χ/Q values for the LOCA, CRDA, and FHA events are presented in Table 3.

The licensee calculated EAB and LPZ atmospheric dispersion factors for the MSLB event using the methodology presented in RG 1.5. This methodology assumes the resulting steam cloud travels downwind at a height of 30 meters and is uniformly distributed in the vertical between the ground and 30 meters (e.g., fumigation conditions). The RG 1.5 methodology also assumes moderately stable atmospheric conditions (stability class F) and a wind speed of 1 meter per second. The resulting EAB and LPZ χ/Q values for the MSLB event are also presented in Table 3.

In conclusion, the NRC staff has 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 χ/Q values are presented in Table 3. On the basis of this review, the NRC staff concludes that the χ/Q values for DBA releases to the LGS EAB and LPZ, as presented in Table 3, are acceptable for use in the DBA CRDAs performed in support of this amendment application.

3.3 TS Changes

Exelon has proposed the following TS changes for LGS Units 1 and 2. The proposed changes apply to both units unless otherwise noted. The NRC staff's review and acceptance of these changes apply to both units unless otherwise noted.

3.3.1 TS Section 1.0, "Definitions”

The proposed change revises the definition of DOSE EQUIVALENT 1-131 in TS definition 1.9 to replace the word "thyroid" with "inhalation committed effective dose equivalent (CEDE)" and to add a reference to "Table 2.1 of Federal Guidance Report 11, Limiting Values of Radio Nuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, ORNL [Oak Ridge National Laboratory], 1989, as described in Regulatory Guide 1.183".

This proposed change is consistent with the TEDE basis of the radiological consequence analyses and provides an improved correlation between the TS specific activity LCO (where this definition is used) and the projected offsite and control room doses. The NRC staff, therefore, finds this proposed change acceptable.

The proposed change also adds the definition of RECENTLY IRRADIATED FUEL as TS definition 1.35. RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours. Subsequent definitions in this section are renumbered to reflect this addition.

The NRC staff has reviewed the proposed change and finds the insertion of this definition and the consequential renumbering of paragraphs to be acceptable.

3.3.2 TS Section 3/4 1.5, "Standby Liquid Control (SLC) System"

The proposed change revises the applicability of TS Section 3.1.5 to include Operational Condition 3 for the SLC system. This change implements AST assumptions regarding the use of the SLC system to control the suppression pool pH following a LOCA involving significant fission product release. Action 3.1.5 has been revised to include action statements for inoperable SLC equipment in Operational Condition 3, which can include going to COLD SHUTDOWN. SR 4.1.5.b.2 is revised to reflect the Boron-10 weight requirement that is equivalent to the current requirements for Sodium Pentaborate at 29 percent enrichment.

The NRC staff finds the proposed changes acceptable and the NRC staff’s evaluation of the SLC system is discussed in Section 3.1.1.2, “LOCA Fission Product Transport”, of this SE.

3.3.3 TS Section Tables 3.3.2-1, "Isolation Actuation Instrumentation Action Statements"

The proposed change revises Table Notation (*) for TS Table 3.3.2-1 by: 1) replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL;" 2) removing "refueling area," since secondary containment can consist of the common refueling area and the reactor enclosure zones; and 3) deleting the "during CORE ALTERATIONS" criteria. The table notation applies to the applicable operation conditions for the LGS refueling area Unit 1 and Unit 2 ventilation exhaust duct radiation - high and the refueling area manual isolation instrumentation. These changes are consistent with TSTF-51.

The NRC staff’s review noted that in its October 10, 2005, supplement, the licensee stated that “this license amendment request is not removing nor changing any function of any

instrumentation associated with the Refueling Area HVAC [heating, ventilation, and air conditioning] system." The licensee explained that the affected instrumentation is associated only with the common refueling area HVAC systems and only provides protection for the refueling area environment in the event of an FHA. The effect of this change is that the automatic isolation capability for the secondary containment provided by the common refueling area instrumentation would not be required for movement of fuel that had decayed for a time greater than "recently irradiated." The licensee changed the applicability to include the entire secondary containment, thus including the reactor enclosure area of a unit being refueled. The reactor enclosure area of an operating unit would still be required to be isolated from the common area by its TS requirements (TS 3.6.5). Deleting the "during CORE ALTERATIONS" criteria is acceptable because the potential dose is bounded by the dose results of the FHA.

Although the automatic isolation function is being removed from the TS, the licensee has committed to (see February 27, 2004, supplement, Attachment 6) NUMARC 93-01, Revision 3, Section 11.3.6.5, "Safety Removal for Removal of Equipment from Service During Shutdown Conditions," which provides for secondary containment closure and control of radioactive release in the event of a fuel-related accident. The approval of this amendment request does not reduce the licensee's responsibility to measure, monitor, or take appropriate actions with respect to radioactive release. The proposed change is consistent with TSTF-51. Therefore, the NRC staff finds that this TS change is acceptable.

3.3.4 TS Section Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements"

The proposed change revises Table Notation (*) for TS Table 4.3.2.1-1 by: 1) replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL;" 2) removing "refueling area," since secondary containment can consist of the common refueling area and the reactor enclosure zones;" and 3) deleting the "during CORE ALTERATIONS" criteria. The table notation applies to the operation conditions for which surveillance is required for the refueling area Unit 1 and Unit 2 ventilation exhaust duct radiation - high and the refueling area manual isolation instrumentation. These changes are consistent with TSTF-51.

The NRC staff finds this SR change to be acceptable based on the review provided above in Section 3.3.3 regarding the change to TS Section Tables 3.3.2-1.

3.3.5 TS Section Table 3.3.7.1-1, "Radiation Monitoring Instrumentation"

The proposed change revises Table Notation (*) for TS Table 3.3.7.1-1 by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and adding the criterion "or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel". The table notation applies to the applicable operation conditions for the main control room normal fresh air supply radiation monitor. In addition, the main control room normal fresh air supply radiation monitor is no longer applicable to Operational Condition 5 and is only required as an alarm function. The trip function is being removed.

The NRC staff's review noted that in its October 10, 2005, supplement, in response to RAI number 28, the licensee acknowledged that the LGS control room is a common facility used to support both Unit 1 and Unit 2. The licensee went on to state that radiation levels are monitored in the control room outside air intake using four separate channels to provide the

appropriate radiation isolation signal and initiate the system isolation actions. The supplement further states that whenever a single unit is operating, the control room isolation system is required to be operational, which includes all four radiation monitors and associated isolation channels to be in full compliance with TS requirements. Although the proposed changes grant some flexibility to a unit in refueling, the TSs of the operating unit would require the radiation monitoring and isolation functions to still be operational and, thus, assure protection for control room operators. The licensee also stated in its response to RAI number 9 that it was no longer pursuing a change in the TS to remove the automatic start feature. Thus, the request to remove the trip function has been withdrawn.

As such, the NRC staff finds that the proposed change to the TS to replace "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and adding the criteria "or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel" to be acceptable. Additionally, removing the applicability for Condition 5 is acceptable.

3.3.6 TS Section Table 4.3.7.1-1, "Radiation Monitoring Instrumentation Surveillance Requirements"

The proposed change revises Table Notation (*) for TS Table 4.3.7.1-1 by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and adding the criterion "or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel". The table notation applies to the operation conditions for which surveillance is required for the main control room normal fresh air supply radiation monitor. In addition, the main control room normal fresh air supply radiation monitor is no longer applicable to Operational Condition 5.

The NRC staff, in its review of TS Section Table 3.3.7.1-1, noted that this instrumentation is common to both units and is required to be operational and monitored with appropriate surveillance if either unit is in an operational (non-shutdown) status. Although the proposed TS change in a refueling unit may allow some flexibility, the TS of an operating unit will still require operability and surveillance of the instrumentation to assure protection to control room operators. The NRC staff finds the change to be acceptable.

3.3.7 TS Section LCO 3.4.7 and SR 4.4.7, "Main Steam Isolation Valves (MSIV)"

The proposed change would have revised LCO 3.4.7 to increase the MSIV maximum closing time from "less than or equal to 5 seconds" to "less than or equal to 10 seconds". Additionally, the proposed change also would have revised SR 4.4.7 to increase the MSIV full closure from "between 3 and 5 seconds" to "between 3 and 10 seconds".

However, the licensee withdrew the proposed change in its October 25, 2004, supplement. As such, the staff did not evaluate this change.

3.3.8 TS 3.6.1.2, Restore Action c., "Primary Containment Leakage"

The proposed change revises the action statement to restore "the leakage rate to <100 scf per hour for any MSIV that exceeds 100 scf per hour." The current restore value is < 11.5 scf per hour, for any MSIV that exceeds 100 scfh, based on the existing radiological analysis.

This change has a direct effect upon dose released during a DBA. The staff finds this change to be acceptable because staff confirmed that the proposed specification change is consistent with the methodology used in the revised DBA dose analyses referenced in Section 3.1.1.5.

3.3.9 TS Section 3.6.5.1.2, “Refueling Area Secondary Containment Integrity”

The proposed change replaces "OPERATIONAL CONDITION *" in the Applicability section of TS 3.6.5.1.2 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability and Action Statements are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting the reference to “CORE ALTERATIONS".

The NRC staff has reviewed the proposed change and finds that the relocation and modification of the Applicability condition is acceptable and consistent with TSTF-51.

3.3.10 TS Section 3.6.5.2.2, "Refueling Area Secondary Containment Automatic Isolation Valves"

The proposed change deletes the "OPERATIONAL CONDITION *" in the Applicability section of TS 3.6.5.2.2 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability and Action Statements are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting reference to “CORE ALTERATIONS".

Although the TSTF-51 program allows certain ESF functions to be inoperable, such as the automatic isolation feature, the ability to isolate the secondary containment in the event of an accident is still required to reduce and mitigate dose. The licensee has committed to meet the objectives of NUMARC 93-01, which directs action to be taken to isolate, monitor and process radioactive effluent in the event of a shutdown accident. The proposed change is consistent with TSTF-51. Therefore, the NRC staff finds that this TS change is acceptable.

3.3.11 TS Section 3.6.5.3, "Standby Gas Treatment System - Common System”

The proposed change deletes the (*) in the Applicability and Action section of TS 3.6.5.3 and the corresponding explanation is relocated from the bottom of the page. Additionally, the Applicability section and Action Statements “a.2" and “b.” are revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting references to "CORE ALTERATIONS". Action Statement “b.” is being revised to only apply to handling of recently irradiated fuel in the secondary containment, or during operations with a potential for draining the reactor vessel.

The NRC staff has reviewed the proposed change and finds that the relocation and modification of the Applicability condition is acceptable and consistent with TSTF-51.

3.3.12 TS Section 4.6.5.3, "Standby Gas Treatment System - Common System"

In its application, dated February 27, 2004, the licensee proposed to increase the charcoal absorber sample acceptance criteria for the methyl iodide penetration tests in SRs 4.6.5.3.b.2 and 4.6.5.3.c from less than 0.5 percent to less than 1.25 percent.

However, the licensee withdrew this proposed change in its October 10, 2005, supplement. As such, the staff did not evaluate this change.

3.3.13 TS Section 4.6.5.4, "Reactor Enclosure Recirculation System"

In its application, dated February 27, 2004, the licensee proposed to relax the following SRs (SR) related to the RERS charcoal absorbers:

* SR 4.6.5.4.a to annotate a flow range through the high-efficiency particulate air (HEPA) filters of a minimum of 30,000 cfm through the HEPA filters

* SR 4.6.5.4.b.1 to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm + 10 percent) instead of annotating a specific flow of 60,000 cfm + 10 percent.

* SR 4.6.5.4.b.2 to verify at least once per 24 months, or (1) after structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following painting, fire, or chemical release in any communicating ventilation zone, that a laboratory analysis of a representative carbon sample obtained shows methyl iodide penetration of less than 15 percent rather than 2.5 percent.

* SR 4.6.5.4.b.3 to verify a subsystem flow rate within a range of 30,000 to 66,000 cfm.

* SR 4.6.5.4.c to verify after 720 hours of operation, that a laboratory analysis of a representative carbon sample shows methyl iodide penetration of less than 15 percent rather than 2.5 percent.

* SR 4.6.5.4.d.1 to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm + 10 percent) instead of annotating a specific flow of 60,000 cfm + 10 percent.

* SR 4.6.5.4.e to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm + 10 percent) instead of annotating a specific flow of 60,000 cfm + 10 percent.

* SR 4.6.5.4.f to clarify that the in-place penetration test is performed at the rated flow rate (60,000 cfm + 10 percent) instead of annotating a specific flow of 60,000 cfm + 10 percent.

However, the licensee withdrew these proposed changes in its October 10, 2005, supplement. As such, the staff did not evaluate these changes.

3.3.14 TS Section 3.7.1.2, "Emergency Service Water System - Common System"

In its application, dated February 27, 2004, the licensee proposed to expand the definition of the (*) to include "handling RECENTLY IRRADIATED FUEL in the secondary containment and during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the LCO, the Applicability section and Action “c.” of TS 3.7.1.2 would

have been deleted and the corresponding explanation relocated from the bottom of the applicable page.

However, the licensee withdrew this proposed change in its October 10, 2005, supplement. As such, the staff did not evaluate this change.

3.3.15 TS Section 3.7.1.3, "Ultimate Heat Sink"

In its application, dated February 27, 2004, the licensee proposed to expand the definition of the (*) to include "handling RECENTLY IRRADIATED FUEL in the secondary containment and during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the reference to OPERATIONAL CONDITION (*) in the Applicability section and Action “c.” of TS 3.7.1.3 would have been deleted and the corresponding explanation relocated from the bottom of the page.

However, the licensee withdrew this proposed change in its October 10, 2005, supplement. As such, the staff did not evaluate this change.

3.3.16 TS Section 3.7.2, "Control Room Emergency Fresh Air Supply System - Common System "

The licensee proposed the following changes to TS 3.7.2:

- Applicability Section - the proposed change expands the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment, or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section is deleted and the corresponding explanation is relocated from the bottom of the page.

- Action “b.” - the operational condition is revised to expand the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment, or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section is deleted and the corresponding explanation is relocated from the bottom of the page.

- Action “b.2" - the action statement is revised by replacing the term "irradiated fuel" with "RECENTLY IRRADIATED FUEL" and deleting reference to "CORE ALTERATIONS".

- Action “c” - the reference to Operational Condition (*) is deleted and the action has been incorporated into Action “b.2.”

- Notation (*) at the bottom of the page is deleted and included in the applicable sections.

The NRC staff reviewed the proposed changes and noted that the LGS control room is a shared system that is governed by both Unit 1 and Unit 2 TSs. As such, even though the propose changes provide some flexibility during outages, if either unit is operating, the TS of the operating unit would be controlling. These changes are consistent with the TSTF-51 agreements and as such, the NRC staff finds the proposed changes to be acceptable.

3.3.17 TS Section 4.7.2, "Control Room Emergency Fresh Air Supply System - Common System"

The licensee proposed changes relaxing the following SRs related to the charcoal absorbers: The charcoal absorber sample acceptance criteria for the methyl iodide penetration tests in SRs 4.7.2.c.2 and 4.2.7.d has been increased from less than 2.5 percent to less than 10 percent. The proposed changes would have revised SR 4.7.2.e.3 to only require verification of the manual initiation of the radiation mode of CREFAS and removes reference to the outside air intake high radiation mode.

However, the licensee withdrew these proposed changes in its October 10, 2005, supplement. As such, the NRC staff did not evaluate this change.

3.3.18 TS Section 3.8.1.2, "AC Sources - Shutdown"

In its application, dated February 27, 2004, the licensee proposed to expand the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.1.2 would have been deleted and the corresponding explanation relocated from the bottom of the page.

However, the licensee withdrew this proposed change in its October 25, 2004, supplement. As such, the staff did not evaluate this change.

3.3.19 TS Section 3.8.2.2, "DC Sources - Shutdown"

In its application, dated February 27, 2004, the licensee proposed to expand the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.2.2 would have been deleted and the corresponding explanation relocated from the bottom of the page. The statement in TS 3.8.2.2 Action “c.” would have been revised to change “irradiated fuel" to "RECENTLY IRRADIATED FUEL.

However, the licensee withdrew these proposed changes in its October 25, 2004, supplement. As such, the staff did not evaluate these changes.

3.3.20 TS Section 3.8.3.2, "Electrical Power Systems, Distribution - Shutdown"

In its application, dated February 27, 2004, the licensee proposed to expand the definition of the (*) to include "when handling RECENTLY IRRADIATED FUEL in the secondary containment or during operations with a potential for draining the reactor vessel" (consistent with TSTF-51). Additionally, the (*) in the Applicability section of TS 3.8.3.2 would have been deleted and the corresponding explanation relocated from the bottom of the page. The statements in TS 3.8.3.2 Actions “a.” and “b.” would have been revised to change “irradiated fuel" to "RECENTLY IRRADIATED FUEL."

However, the licensee withdrew these proposed changes in its October 25, 2004, supplement. As such, the NRC staff did not evaluate these changes.

3.3.21 TS Changes Summary

Proposed TS changes to the following sections have been reviewed by the NRC staff and found to be acceptable:

TS Section 1.0, "Definitions"
TS Section 3/4 1.5, "Standby Liquid Control (SLC) System"
TS Section Tables 3.3.2-1, "Isolation Actuation Instrumentation Action Statements"
TS Section Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements"
TS Section Table 3.3.7.1-1, "Radiation Monitoring Instrumentation"
TS Section Table 4.3.7.1-1, "Radiation Monitoring Instrumentation Surveillance Requirements"
TS 3.6.1.2, Restore Action c., "Primary Containment Leakage"
TS Section 3.6.5.1.2, "Refueling Area Secondary Containment Integrity"
TS Section 3.6.5.2.2, "Refueling Area Secondary Containment Automatic Isolation Valves"
TS Section 3.6.5.3, "Standby Gas Treatment System - Common System"

Implementation of these TS changes has minimal effect on control room operator dose and the release of contamination to the public. As such, there is negligible impact on public health and safety.

During the NRC staff's review, the licensee withdrew changes to the following TS sections. These changes were not reviewed, therefore, no finding of acceptability was made.

TS Section 3.4.7 and 4.4.7, "Main Steam Isolation Valves (MSIV)"
TS Section 4.6.5.3, "Standby Gas Treatment System - Common System"
TS Section 4.6.5.4, "Reactor Enclosure Recirculation System"
TS Section 3.7.1.2, "Emergency Service Water System - Common System"
TS Section 3.7.1.3, "Ultimate Heat Sink"
TS Section 3.7.2, "Control Room Emergency Fresh Air Supply System - Common System "
TS Section 3.8.1.2, "AC Sources - Shutdown"
TS Section 3.8.2.2, "DC Sources - Shutdown"
TS Section 3.8.3.2 "Electrical Power Systems, Distribution - Shutdown"

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (70 FR 29794). 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.

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Table 1

Limerick Accident Analysis Parameters

General

Reactor power (3458 x 1.02) megawatts thermal (MWt)	3,527
Design core burnup, effective full power days	711
Core inventory, Ci/MWt	Submittal Attachment 8, Table 1[1]
Dose conversion factors	FGR11/FGR12
Breathing rates, offsite, cubic meters per second (m^3/s)	
0-8 hours	3.5E-4
8-24 hours	1.8E-4
>24 hours	2.3E-4
Breathing rate, control room, m^3/s	3.5E-4
Unfiltered inleakage due to ingress and egress, cubic feet per minute (cfm)	10
Control room unfiltered infiltration in radiation isolation mode, cfm[2]	225
Control room normal intake flow, cfm[3]	2,200
Control room filtered pressurization maximum flowrate, cfm	525
Control room filtered recirculation, cfm	2,175
Control room volume, cubic feet (ft^3)	126,000
Control room intake filter efficiency, percent	
Particulates	99
Elemental and organic iodine	95
Noble gases	0
Control room recirculation filter efficiency, percent	
Particulates	99
Elemental and organic iodine	95
Noble gases	0

[1] Bounding isotopic activity is used.

[2] Includes 10 cfm unfiltered inleakage for ingress and egress. The 10 cfm unfiltered inleakage is added to the measured unfiltered inleakage to determine the total unfiltered inleakage.

[3] For the FHA and CRDA, emergency pressurization is not credited. The total unfiltered inleakage is assumed to be 100,000 cfm, which includes 10 cfm for ingress and egress. For the MSLB, the dose is taken at the control room intake and is not dependent upon the flow into the control room.

Table 1
Limerick Accident Analysis Parameters (Cont.)

General (Cont.)

Control room occupancy factor	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4
Control room atmospheric dispersion values (χ/Q values), sec/m^3	Table 2
Offsite χ/Q values, sec/m^3	Table 3

Loss-of-Coolant Accident (LOCA)

Containment Leakage Source

Onset of gap release phase, min	2.0

Core release fractions and timing– to containment atmosphere

Duration, hrs	0.5	1.5
Noble Gases	0.05	0.95
Halogens	0.05	0.25
Alkali Metals	0.05	0.20
Tellurium	0.00	0.05
Strontium	0.00	0.02
Barium	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

Iodine species distribution	
Aerosol	0.95
Elemental	0.0485
Organic	0.0015
Primary containment volume, ft^3	
Drywell	231,401
Minimum suppression pool air space	147,670
Minimum suppression pool water volume	118,655
Containment leak rate, percent/day	
0- 24 hours	0.5
Greater than 24 hours	0.25
SGTS filter effective efficiency, percent	
Before drawdown (all species)	0
After drawdown (all species except noble gases)	99
After drawdown (noble gases)	0

Table 1

Limerick Accident Analysis Parameters (Cont.)

LOCA (Cont.)

Secondary containment bypass (percent primary containment volume/day)	0.00
Secondary containment (reactor enclosure) volume[4], ft^3	1.8E6
SGTS drawdown time, min	15.5
Reactor enclosure recirculation system (RERS) initiation time, min	3.5
RERS filter effective efficiency, percent	
Before drawdown (all species)	0
After drawdown (particulates)	99
After drawdown (elemental and organic iodine)	95
Effective credit for mixing, percent of reactor enclosure volume	
3.5 - 15.5 minutes	6.98
15.5 min - 30 days	50
Control room isolation delay, minutes	0
MSIV Leakage	
MSIV TS leak rate[5] at test pressure of $22 pounds per square inch gauge, standard cubic feet per hour	
One line	100
Total	200
Normal steamline (and steam) temperature, EF	550.0
ESF Leakage	
Iodine species, percent	
Particulate/aerosol	0
Elemental	97
Organic	3
Iodine flash fraction	0.1
ESF estimated leakage into secondary containment, gpm	5[6]

4 A maximum of 50 percent of this volume is credited for mixing. The SGTS was assumed to have a flow rate of 3000 cfm during the drawdown and 2500 cfm after the drawdown.

5 MSIV leakage is reduced 55.1 percent after 24 hours.

6 The allowable ESF leakage limit (2.5 gpm) is doubled per RG 1.183. The allowable ESF leakage value is not corrected for accident conditions. Therefore, the ESF leakage limit is assumed to be at accident conditions and is not the limit for normal operating conditions.

Table 1

Limerick Accident Analysis Parameters (Cont.)

Fuel Handling Accident

Peaking factor	1.7
Fuel rods damaged, rods[7]	212
Decay period, hrs	24
Pool decontamination factor	
Iodine, effective	200
Noble gases	1
Particulate	Infinite
Fraction of core inventory in gap	
I-131	0.08
Kr-85	0.10
Other iodines	0.05
Other noble gases	0.05
Alkali metals[8]	0.12
Release period, hr	2
Release location	
SGTS operating	SGTS north stack
No SGTS operation	Reactor building south stack[9]
CREFAS initiation	Not credited
Control room unfiltered intake, cfm	126,000
Water depth, ft	$23

[7] Based upon fuel assembly with an 8x8 fuel pin array.

[8] Cesium and rubidium are present and were considered, but were not included in the calculation because the decontamination factor for particulate is assumed to be infinite.

[9] The exhaust point under the assumed no filtration condition is the reactor building south stack. The north stack, which is used for releases filtered by the SGTS, is located closer to the control room intake and, therefore, has a higher dispersion factor than the south stack. However, the SGTS is designed to remove at least 99 percent of the iodine that would otherwise be released; this filtration more than overcomes the effect of the higher dispersion factor, so the south stack release unfiltered is bounding.

Table 1

Limerick Accident Analysis Parameters (Cont.)

MSLB

Liquid coolant release discharged mass, lb	140,000
MSIV closure time, sec	5.5
Reactor coolant system pressure, pounds per square inch absolute	1060
Reactor coolant system temperature, EF	548
Reactor coolant activity[10], µCi/gm dose equivalent I-131	
Normal	0.2
Spike	4.0
Radioactivity release rate to environment	Instantaneous
Control room occupancy factor	1
CREFAS initiation	Not credited

CRDA

Peaking factor	1.7
Fraction of core inventory in gap	
Noble gases	0.1
Iodine	0.1
Other halogens	0.05
Alkali metals	0.12
Amount of core with damaged fuel rods, percent	1.8
Damaged rods that fail, percent	0.77
Melted fuel release fraction to vessel	
Noble gases	1.0
Iodine	0.5
Br	0.3
Alkali metals	0.25
Tellurium metals	0.05
Ba, Sr	0.02
Noble metals	0.0025
Ce	0.0005
La	0.0002

[10] Cesium in the reactor coolant is also considered.

Table 1

Limerick Accident Analysis Parameters (Cont.)

CRDA (Cont.)

Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
Others	0.01
Fraction of activity released from main condenser	
Noble gases	1.0
Iodine[11]	0.1
Others	0.01
Release rate from main condenser, percent/day	1
Release duration, hours	24
CREFAS initiation	Not credited
Control room unfiltered intake, cfm	126,000

[11] Iodine released is 97 percent elemental and 3 percent organic.

Table 2

Limerick Units 1 & 2

Control Room Atmospheric Dispersion Factors

Release Pathway	Accidents[12]	χ/Q Value (sec/m^3)				
		0–2 hr	2–8 hr	8–24 hr	1–4 day	4–30 day
North Stack	LOCA, CRDA	6.88×10^{-3}	5.17×10^{-3}	2.04×10^{-3}	1.29×10^{-3}	9.63×10^{-4}
South Stack	FHA[13]	1.26×10^{-3}	--	--	--	--

Table 3

Limerick Units 1 & 2

Offsite Atmospheric Dispersion Factors

Receptor	Accidents	χ/Q Value (sec/m^3)					
		Puff[14]	0–2 hr	0–8 hr	8–24 hr	1–4 day	4–30 day
EAB	LOCA, CRDA, FHA	--	3.18×10^{-4}	--	--	--	--
	MSLB	4.77×10^{-4}	--	--	--	--	--
LPZ	LOCA, CRDA, FHA	--	1.15×10^{-4}	5.79×10^{-5}	4.10×10^{-5}	1.95×10^{-5}	6.68×10^{-6}
	MSLB	1.89×10^{-4}	--	--	--	–	--

12 Control room χ/Q values were not calculated for the MSLB event. MSLB doses to the control room were calculated assuming a hemispheric radioactive steam cloud is transported without dilution over the control room air intake location for the duration required for the cloud to pass by the intake assuming a wind speed of 1 meter per second.

13 Use of the South Stack χ/Q values to model FHA releases is contingent upon requiring SGTS operability when moving irradiated fuel, performing core alterations, or during operations with the potential to drain the vessel.

14 The MSLB event was modeled as an instantaneous (e.g., puff) release to the environment.