September 11, 2006

Mr. Christopher M. Crane, President
and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADOPTION OF ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NOS. MB6530, MB6531, MB6532, MB6533, MC8275, MC8276, MC8277 AND MC8278)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 221 to Renewed Facility Operating License No. DPR-19 and Amendment No. 212 to Renewed Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3, and Amendment No. 233 to Renewed Facility Operating License No. DPR-29 and Amendment No. 229 to Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated October 10, 2002, as supplemented by letters dated March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 2006.

The amendments adopt the alternative source term methodology by replacing the current accident source term described in Technical Information Document 14844 with an accident source term as prescribed in Title 10 to the *Code of Federal Regulations* Section 50.67. The amendments also revise the technical specification sections associated with the implementation of Technical Specification Task Force (TSTF) - 51 traveler, which provides some relaxation of certain requirements during movement of irradiated fuel.

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

Sincerely,

/RA/

Maitri Banerjee, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, and 50-265

Enclosures:
1. Amendment No. 221 to DPR-19
2. Amendment No. 212 to DPR-25
3. Amendment No. 233 to DPR-29
4. Amendment No. 229 to DPR-30
5. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

/RA/

Maitri Banerjee, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, and 50-265

Enclosures:
1. Amendment No. 221 to DPR-19
2. Amendment No. 212 to DPR-25
3. Amendment No. 233 to DPR-29
4. Amendment No. 229 to DPR-30
5. Safety Evaluation

cc w/encls: See next page

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OFFICE	LPL3-2/PM	LPL3-2/LA	DRA/ACVB/BC	DCI/CSGB/BC	DE/EEEB/BC
NAME	MBanerjee:mw	DClarke	RDennig	AHiser	GWilson
DATE	9/12/06	8/25/2006	8/17/06	8/16/06 (w/edit.)	8/14/06

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NAME	MKotzalas	NSalgado	TKobetz	Ahodgdon (w/comm.)	DCollins
DATE	8/14/06	08/14/06	08/18/06	9/08/2006	9/11/2006

OFFICIAL RECORD COPY

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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 221
Renewed License No. DPR-19

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated October 10, 2002, as supplemented by letters dated March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 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 Renewed Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: September 11, 2006

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 212
Renewed License No. DPR-25

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated October 10, 2002, as supplemented by letters dated March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 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 3.B. of Renewed Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: September 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NOS. 221 AND 212

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

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

REMOVE	INSERT
Unit 2 License Page 3	Unit 2 License Page 3
Unit 3 License Page 4	Unit 3 License Page 4
1.1-2	1.1-2
1.1-3	1.1-3
3.1.7-1	3.1.7-1
3.3.6.1-7	3.3.6.1-7
3.3.6.2-4	3.3.6.2-4
3.3.7.1-1	3.3.7.1-1
3.6.1.3-9	3.6.1.3-9
3.6.4.1-1	3.6.4.1-1
3.6.4.1-2	3.6.4.1-2
3.6.4.2-1	3.6.4.2-1
3.6.4.2-3	3.6.4.2-3
3.6.4.3-1	3.6.4.3-1
3.6.4.3-2	3.6.4.3-2
3.6.4.3-3	3.6.4.3-3
3.7.4-1	3.7.4-1
3.7.4-2	3.7.4-2
3.7.5-1	3.7.5-1
3.7.5-2	3.7.5-2
3.8.1-15	3.8.1-15
3.8.2-1	3.8.2-1
3.8.2-3	3.8.2-3
3.8.2-4	3.8.2-4
3.8.5-1	3.8.5-1
3.8.8-1	3.8.8-1
3.8.8-2	3.8.8-2
5.5-12	5.5-12

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
License No. DPR-29

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated October 10, 2002, as supplemented by letters dated March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 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 3.B. of Renewed Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: September 11, 2006

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 229
License No. DPR-30

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

   A. The application for amendment by Exelon Generation Company, LLC, et al. (the licensee) dated October 10, 2002, as supplemented by letters dated March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 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 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 229, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Daniel S. Collins, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Renewed Facility Operating License

Date of Issuance: September 11, 2006

ATTACHMENT TO LICENSE AMENDMENT NOS. 233 AND 229

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

DOCKET NOS. 50-254 AND 50-265

Replace the following pages of the Facility Operating Licenses and Appendix “A” Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Unit 1 License Page 4	Unit 1 License Page 4
Unit 2 License Page 4	Unit 2 License Page 4
1.1-2	1.1-2
1.1-3	1.1-3
3.1.7-1	3.1.7-1
3.3.6.1-7	3.3.6.1-7
3.3.6.2-4	3.3.6.2-4
3.3.7.1-4	3.3.7.1-4
3.6.1.3-8	3.6.1.3-8
3.6.4.1-1	3.6.4.1-1
3.6.4.1-2	3.6.4.1-2
3.6.4.2-1	3.6.4.2-1
3.6.4.2-3	3.6.4.2-3
3.6.4.3-1	3.6.4.3-1
3.6.4.3-2	3.6.4.3-2
3.6.4.3-3	3.6.4.3-3
3.7.4-1	3.7.4-1
3.7.4-2	3.7.4-2
3.7.5-1	3.7.5-1
3.7.5-2	3.7.5-2
3.8.1-15	3.8.1-15
3.8.2-1	3.8.2-1
3.8.2-3	3.8.2-3
3.8.2-4	3.8.2-4
3.8.5-1	3.8.5-1
3.8.8-1	3.8.8-1
3.8.8-2	3.8.8-2
5.5-12	5.5-12

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED

TO AMENDMENT NO. 221 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 212 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 233 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 229 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated October 10, 2002 (Reference 1), as supplemented by letters dated March 21 (Reference 2), March 28 (Reference 3), August 4 (Reference 4), September 15 (Reference 5) and October 31, 2003 (Reference 6), and June 30 (Reference 7), August 6 (Reference 8), September 3 (Reference 9), September 10 (Reference 10), September 22 (Reference 11), November 2 (Reference 12) and November 5, 2004 (Reference 13), and March 3 (Reference 14), August 22 (Reference 15), September 3 (Reference 16) and September 27, 2005 (Reference 17), and February 17 (Reference 18) and May 25, 2006 (Reference 19), Exelon Generation Company, LLC, et al. (Exelon, the licensee) requested changes to the technical specifications (TSs) and surveillance requirements (SRs) for Dresden Nuclear Power Station, Units 2 and 3 (Dresden), and Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities). The proposed changes would adopt the alternative source term (AST) methodology by replacing the current accident source term described in Technical Information Document (TID) 14844 (source term) with an accident source term as prescribed in Title 10 to the *Code of Federal Regulations* (10 CFR) Section 50.67, “Accident source term.” The submittals also proposed to revise the TS sections associated with the implementation of Technical Specification Task Force Traveler-51 (TSTF-51), “Revised Containment Requirements During Handling Irradiated Fuel and Core Alteration,” which provides some relaxation of certain requirements during refueling operations concerning the control room emergency ventilation (CREV), standby gas treatment (SGT), and secondary containment systems.

The March 21, March 28, August 4, September 15 and October 31, 2003, and June 30, August 6, September 3, September 10, September 22, November 2 and November 5, 2004, and March 3, August 22, September 3 and September 27, 2005, and February 17 and May 25, 2006, supplements contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

To support the proposed full implementation of an AST, Exelon re-analyzed the radiological consequences of the following four design-basis accidents (DBAs):

1. Large Break Loss-of-Coolant Accident (LBLOCA)
2. Main Steam Line Break (MSLB)
3. Control Rod Drop Accident (CRDA)
4. Fuel-Handling Accident (FHA)

The technical evaluation of the above four DBAs implementing an AST is provided in Section 3.1, "Radiological Consequences of Design-Basis Accidents" of this Safety Evaluation (SE).

Also, as part of the AST implementation, Exelon requested changes to the following Dresden and Quad Cities TSs:

5. TSs associated with the implementation of TSTF-51 traveler which provides for relaxation of certain requirements during refueling operations

   - Secondary Containment Isolation Instrumentation, Table 3.3.6.2-1, Note b
   - Control Room Emergency Ventilation (CREV) System Isolation Instrumentation, Table 3.3.7.1-1, Note b, Quad Cities only
   - Control Room Emergency Ventilation (CREV) Instrumentation, 3.3.7.1 Applicability, Dresden only
   - Secondary Containment, 3.6.4.1 Applicability and Action C
   - Secondary Containment Isolation Valves (SCIVs), 3.6.4.2 Applicability and Action D
   - Standby Gas Treatment (SGT) System, 3.6.4.3 Applicability and Actions C and F
   - Control Room Emergency Ventilation (CREV) System, 3.7.4 Applicability and Action C
   - Control Room Emergency Ventilation Air Conditioning (AC) System, 3.7.5 Applicability and Action C
   - AC [Alternating Current] Sources - Operating, Surveillance Requirement 3.8.1.21
   - AC [Alternating Current] Sources - Shutdown, 3.8.2 Applicability and Actions A and B
   - DC [Direct Current] Sources - Shutdown, 3.8.5 Applicability and Action A
   - Distribution Systems - Shutdown, 3.8.8 Applicability and Action A

2. Standby Liquid Control (SLC) System, 3.1.7 Applicability and Action C
3. Primary Containment Isolation Instrumentation, Table 3.3.6.1-1
4. Ventilation Filter Test Program Sections 5.5.7.c and e, and Standby Gas Treatment Surveillance SR 3.6.4.3.1 (these changes have been withdrawn)
5. Definitions, Section 1.1, Dose Equivalent I-131
6. Primary Containment Leakage Rate Testing Program Section 5.5.12.c
7. Main Steam Isolation Valve (MSIV) Leakage Surveillance SR 3.6.1.3.10

Subsequently, Exelon modified some of the proposed TS changes originally submitted by letter dated October 10, 2002, by the following supplemental letters:

- September 15, 2003 - Updated the definition of Dose Equivalent I-131 and withdrew proposed TS changes in Item 4 above,
- September 22, 2004 - Re-updated the definition of Dose Equivalent I-131, superceding the change identified in the September 15, 2003, submittal above, and
- August 22, 2005 - Revised the MSIV leakage rate acceptance criteria in SR 3.6.1.3.10.

2.0 REGULATORY EVALUATION

The current radiological consequence analyses for the DBAs for Dresden and Quad Cities are based on the TID-14844 source term. In this license amendment, the licensee requested a full-scope implementation of the AST, as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and pursuant to 10 CFR 50.67. The use of an AST is addressed in 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional source term used in their DBA radiological consequence analyses.

The NRC staff evaluated the radiological consequences of affected DBAs against the dose criteria specified in 10 CFR 50.67; these criteria are 25 rem, total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the entire period of passage of the postulated release, and 5 rem TEDE in the control room (CR) for the duration of the accident.

The NRC staff used applicable guidance in Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term," and RG 1.183. Other relevant regulatory documents used by the NRC staff are General Design Criterion (GDC) 19, "Control room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, and 10 CFR 50.36, "Technical Specifications," as it relates to criteria for including an item as a limiting condition for operation in TSs.

This SE addresses (1) the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results, and (2) acceptability of the proposed TS changes. The regulatory requirements upon which the NRC staff based its acceptance are the accident dose criteria in 10 CFR. 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, SRP 15.0.1, and GDC 19. Except where the licensee has proposed a suitable alternative to regulatory guidance, the NRC staff used the regulatory requirements and guidance in the following documents in its review of the requested amendment and the radiological consequence dose analyses:

- 10 CFR 50.36
- 10 CFR 50.67
- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"
- 10 CFR 50.90, "Application for amendment of license or construction permit;" and 10 CFR 50.92, "Issuance of amendment"

- GDC 19 of Appendix A to 10 CFR Part 50, and GDC 60, "Control of releases of radioactive materials to the environment”
- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”
- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”
- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors”
- RG 1.23, “Onsite Meteorological Programs”
- RG 1.5, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors.”
- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”
- NUREG-0800, “Standard Review Plan,” Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”
- NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability Systems” (with regard to control room meteorology)
- NUREG-0800, "Standard Review Plan,” Section 9.4.5, "Engineered Safety Feature Ventilation System”
- NUREG-0800, “Standard Review Plan,” Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Term”
- NUREG-0933, Issue 187, “The Potential Impact of Postulated Cesium Concentration on Equipment Qualification.”
- NUREG-1465, “Accident Source Terms for Light-Water Nuclear Reactor Plants”
- NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containment”

The NRC staff also considered the following regulatory guidance in evaluating the licensee’s proposed TS changes involving the implementation of TS TSFT-51, which provides relaxation of some requirements during refueling operations.

- TSTF-51, Revision 2. Approved by the NRC on October 13, 1999.

- The model TS contained in the improved standard technical specification (ISTS), NUREG-1433, Revision 2, “Standard Technical Specifications, General Electric Plants, BWR/4,” dated April 2001.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design Basis Accidents

To demonstrate that the performance of various plant safety systems designed to mitigate the postulated radiological consequences at Dresden and Quad Cities will remain adequate after implementing the AST and the TS changes requested in this license amendment request (LAR), Exelon re-analyzed the radiological consequences of the following four DBAs:

- LBLOCA
- MSLB

- CRDA
- FHA

The licensee's submittal and supplements provided the results of the radiological consequence analyses for the above DBAs to show compliance with 10 CFR 50.67 for offsite doses and 10 CFR Part 50, Appendix A, GDC 19 for the CR dose.

In its radiological analyses of DBAs, Exelon used an NRC radiological consequence computer code, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.02, as described in NUREG/CR-6604, "A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.02. The RADTRAD code, developed by the Sandia National Laboratories for NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff reviewed and verified the RADTRAD code input and output files provided by Exelon and confirmed that the input parameters and assumptions are consistent as described in the proposed LAR. The NRC staff performed an independent radiological consequence analysis to confirm the results of the offsite and CR doses provided by Exelon.

The parameters and assumptions used for the radiological consequence analyses for Dresden and Quad Cities are the same except for the main CR and the main condenser volumes (see Tables 3 and 4), and their respective offsite and CR atmospheric dispersion factors (see Section 3.2, "Atmospheric Dispersion Estimates," and Tables 7 through 14). Where there are differences in the main steam line configuration and its layout, Exelon used the more conservative parameters (shorter pipe length and smaller volumes for less aerosol deposition) of Quad Cities (instead of Dresden) for both the Dresden and Quad Cities radiological consequence analyses.

3.1.1 Loss-of-Coolant Accident (LOCA)

The current radiological consequence analysis for the postulated LOCA is based on the accident source term described in TID-14844 and it is described in Dresden and Quad Cities Updated Final Safety Analysis Reports (UFSARs) Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Piping Breaks Inside Containment." To demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at Dresden and Quad Cities will remain adequate after this implementation of the AST and the TS changes requested in this LAR, the licensee re-analyzed the offsite and CR radiological consequences of the postulated LOCA.

The licensee calculated and submitted the results of its offsite and CR doses and provided the major assumptions and parameters used in its dose calculations. As documented in its submittals, the licensee has determined that after implementation of the TS changes requested in this LAR and use of an AST, the existing ESF systems at Dresden and Quad Cities will still provide reasonable assurance that the radiological consequences of the postulated LOCA at the EAB, in the LPZ, and in the CR will meet the acceptable radiation dose criteria specified in 10 CFR 50.67. As part of the implementation of the AST, the TEDE acceptance criteria of 10 CFR 50.67(b)(2) replace the previous whole-body and thyroid dose guidelines of 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and GDC 19.

Exelon evaluated the offsite and CR radiological consequences due to the design-basis LOCA.

In accordance with RG 1.183 guidance, Exelon determined the inventory of fission products in the reactor core based on the maximum full power operation of the core (3016 megawatts thermal (MWt), which is 102 percent of the 2957 MWt-rated power) using an appropriate isotope generation and depletion computer code. Fission products from the damaged fuel are released into the reactor coolant system (RCS) and then into the primary containment (i.e., drywell and wetwell). With the postulated LOCA, it is anticipated that the initial fission product release to the primary containment will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins 2 minutes after the event starts and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase begins. The in-vessel release phase continues for 1.5 hours. Tables 1, 4, and 5 of RG 1.183 define the source term used for these two phases.

The inventory in each phase is released at a constant rate over the duration of the phase. Once dispersed in the primary containment, the release to the environment is assumed to occur through the following three pathways:

- leakage of primary containment atmosphere
- leakage of primary containment atmosphere through MSIVs
- leakage from emergency core cooling systems (ECCS) that recirculate suppression pool water outside of the primary containment

Dresden and Quad Cities do not routinely purge the containment during power operation to relieve containment pressure or reduce containment hydrogen concentration following the postulated LOCA. Therefore, the release from containment purge is not analyzed.

3.1.1.1 Primary Containment Atmosphere Leakage

The LOCA considered in this evaluation is a complete and instantaneous circumferential severance of one of the recirculation loops, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. 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 down through the torus downcomer vent pipes and into the suppression pool water, thereby condensing the steam and reducing the drywell pressure. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release occurs in phases over a 2-hour period.

Exelon assumed that the fission products released from the RCS following the postulated LOCA are instantaneously and homogeneously mixed throughout the free air volume of the drywell for the first 2 hours. As characterized in NUREG-1465, the fission product releases (gap and early-in-vessel releases) terminate 2 hours after the onset of the postulated LOCA. This would require reflooding of the RPV. Exelon asserts, and the NRC staff agrees, that reflooding and core quenching would occur at about 2 hours resulting in substantial steam production in the RPV that will purge a large fraction of the drywell atmosphere through the drywell main vent, the main vent header, eight torus vent pipes, 96 downcomers arranged in pairs and submerged in the suppression pool water, the suppression pool water, wetwell air space, and finally back to the drywell through vacuum breaker lines.
The NRC staff believes that the mass and energy (steaming and steam condensation) created by reflooding (arresting RPV failure) and core quenching will provide sufficient energy to mix the

drywell and wetwell air when vacuum breaker cycling occurs during this pressure transient. The licensee assumed that the radioactivity release is diluted into the larger volume of the wetwell plus drywell air spaces after 2 hours. Before this time, the radioactivity is only assumed to be released into the drywell net free volume. The NRC staff expects that most fission products (other than noble gases and iodine in organic form) in the drywell air transferred to the torus air space after 2 hours will be scrubbed by the suppression pool water. However, Exelon did not credit any reduction in fission products transferred to the torus air space from drywell by suppression pool scrubbing. Instead, Exelon assumed a well-mixed torus air space and drywell after 2 hours.

The well-mixed assumption of torus air space and drywell air is precedented in the AST license amendments for Hope Creek Generating Station (ADAMS Accession Number ML012600176), Fermi 2 (ADAMS Accession Number ML042780426), Browns Ferry Nuclear Plant Units 1, 2, and 3 (ADAMS Accession Number ML043100345), Vermont Yankee Nuclear Power Station (ADAMS Accession Number ML040980062), and Brunswick Steam Electric Plants (ADAMS Accession Number ML060540234). Only Duane Arnold Energy Center elected not to assume a well-mixed torus air space and drywell (ADAMS Accession Number ML011660142). They are all boiling-water reactors (BWRs) with Mark I containment design.

The fission products in the containment atmosphere following the postulated LOCA at Dresden and Quad Cities are mitigated by natural deposition of fission products in aerosol form and by the removal of elemental iodine by deposition on surfaces inside the drywell. Consistent with the guideline provided in NUREG/CR-6189, the licensee credited the removal of fission products in aerosol form by natural deposition in the drywell following the postulated LOCA using the Powers simplified model in the RADTRAD, Version 3.02. The licensee used a 10-percentile confidence interval (90-percent probability) value in the RADTRAD code. This simplified model in NUREG/CR-6189 was derived by correlation of results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff finds that the use of this model in NRC computer code, RADTRAD, is acceptable.

Consistent with the guideline provided in SRP 6.5.2 and RG 1.183, the licensee credited the removal of elemental iodine by deposition on surfaces inside the drywell and using the RADTRAD code. The maximum elemental iodine decontamination factor is limited to 200 and it occurs at 3.05 hours into the postulated LOCA. The NRC staff finds this credit to be acceptable. The Exelon evaluation took no credit for removal of iodine by the drywell sprays.

The current design-basis maximum allowable containment leak rates (L_a) in the Dresden and Quad Cities TSs are 1.0 and 1.6 percent by weight of the containment atmosphere air mass per day (percent per day) at a design-basis LOCA maximum containment pressure of 48 pounds per square inch gauge (psig) for Dresden and Quad Cities, respectively. In this LAR, the licensee requested to increase the maximum allowable containment leak rate to 3 percent per day for both Dresden and Quad Cities. The licensee conservatively elected not to claim a reduction of the maximum allowable containment leak rate after 24 hours. Exelon and the NRC staff used the maximum allowable containment leak rate of 3 percent per day for the entire duration (30 days) of the postulated LOCA in their respective dose calculations.

Each single reactor building at Dresden and Quad Cities completely encloses its reactors and primary containment of both units. The reactor building ventilation system maintains the reactor building atmosphere at a slight negative pressure during plant normal operation. Following the postulated LOCA, the fission products in the containment atmosphere are further mitigated by dilution in the reactor building (secondary containment) and by the SGT system filtration. Consistent with the guidance provided in RG 1.183, the licensee assumed that leakage from the primary containment would mix with the reactor building free air with no more than 50 percent mixing efficiency.

The SGT system is a safety-related system and consists of two redundant trains. Each train consists of, among other things, charcoal filter and pre and post high-efficiency particulate air (HEPA) filters. The charcoal adsorbers consist of a 2-inch thick layer of activated carbon impregnated with potassium iodide. In-place penetration and bypass testing of HEPA filters and charcoal absorbers, and laboratory testing of charcoal samples are specified in TS 5.5.7, "Ventilation Filter Testing Program." This testing is in accordance with the guidance provided in RG 1.52 (Revision 2), American National Standards Institute/American Society of Mechanical Engineers N510 (1980), and American Society for Testing and Materials D3803 (1989). With the requirements specified in TSs, the licensee conservatively assumed an organic and elemental iodine removal efficiency of 50 percent (instead of 97.5 percent entitled by the required tests) by the charcoal absorber and a particulate removal efficiency of 99 percent by HEPA.

Exelon has taken no exception or departure from the guidance provided in RG 1.183 in evaluating the radiological consequence resulting from this fission product release pathway.

3.1.1.2 MSIV Leakage

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

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

Exelon assumed that the outboard MSIVs are closed on all four main steam lines and one inboard MSIV in the broken line is open. The licensee assumed a maximum MSIV leakage of 60 standard cubic feet per hour (scfh) in the broken line, two unbroken lines are assumed to leak at 60 and 30 scfh, and the third unbroken line is assumed not to leak. These leak rates (a total of 150 scfh) are based on a design-basis LOCA maximum peak containment pressure of 48 psig. Exelon did not credit any reduction in drywell pressure or the MSIV leakage rate of 150 scfh after 24 hours following the postulated LOCA. Leakage rates were held constant for the entire duration of the accident (30 days) for conservatism.

The licensee's analysis does take credit for aerosol and iodine removal in the main steam lines. The licensee's iodine removal modeling assumes conservative well-mixed control volumes. Only the volumes associated with horizontal runs of seismically qualified main steam line piping are included in the modeling of aerosol deposition. The licensee assumes two aerosol settling volumes (nodes) for two unbroken main steam lines; one node between the RPV and the inboard MSIV, and the other node between the inboard and outboard MSIVs. The licensee conservatively assumed that the broken main steam line does not have the volume between the RPV and inboard MSIV available for iodine or aerosol removal, thus assuming only one aerosol settling node between the inboard and outboard MSIVs. The licensee's main steam line modeling is conservative because it minimizes aerosol deposition credit.

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 (Perry) (ADAMS Accession Number ML021840462) and Clinton Nuclear Station (Clinton) (ADAMS Accession Number ML052570461). The aerosol settling model is described in a report, AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," which was prepared by the NRC Office of Nuclear Regulatory Research. AEB-98-03 gives a distribution of aerosol settling velocities that are estimated to apply in the main steam line piping. The model used in the Perry and Clinton assessments assumed aerosol settling may occur in the main steam lines upstream of the outboard MSIV, at the median (50th percentile) settling velocity given by the Monte Carlo analysis described in the AEB-98-03 report. Exelon used the same aerosol settling model as described in AEB-98-03, but used the 40th percentile settling velocity for Dresden and Quad Cities to be more conservative. Exelon assumed, as approved for Perry and Clinton by the NRC staff, that settling would occur in one settling volume between the inboard MSIV and the outboard MSIV for the main steam line, which has been assumed broken inside the drywell. For the remaining two unbroken main steam lines, aerosol settling is assumed to occur in two settling volumes, one between the reactor vessel and the inboard MSIV, the other volume between the two closed MSIVs. Exelon used only the horizontal pipe projected area (pipe diameter times length) for determining the aerosol deposition rates and removal efficiencies.

The NRC staff expressed a concern that the removal through aerosol settling was overestimated by modeling two settling volumes with the same settling velocity in each, when the settling would be expected to be at a lesser rate for the later sections of piping and at a later time considering that the larger and heavier aerosols would have already settled out of the main steam line atmosphere in upstream sections of piping. However, as stated above, Exelon did not credit any reduction in drywell pressure or the MSIV leakage rate after 24 hours. Leakage rates were assumed to be held constant for the entire duration of the accident for conservatism. Given this information, the NRC staff finds the Dresden and Quad Cities main steam line aerosol settling model to be reasonably conservative.

The NRC staff acknowledges that aerosol settling is expected to occur in the main steam line piping but because of recent concerns with aerosol sampling and its characteristics used in AEB-98-03 and lack of further information, the NRC staff is concerned with how much deposition (i.e., what settling velocity value) is appropriate. The licensee has used a model based on the methodology of AEB-98-03, but has applied additional conservatism (i.e., 40th percentile settling velocity, constant MSIV leakage for the entire duration of the accident) to address the NRC staff's concern about the applicability of the AEB-98-03 methodology to Dresden and Quad Cities. The NRC staff further acknowledges that the estimate of the fraction of the aerosol that leaks to the environment is uncertain because of phenomenological uncertainties concerning the environment the aerosol encounters in the various volumes assumed by Exelon. The NRC staff sought a qualitative depiction of this uncertainty by considering that (1) the system is isothermal and isobaric, (2) only gravitational deposition of aerosol is considered, (3) agglomeration of the aerosol is neglected, (4) leak rates are constant, and (5) the gas phase within each volume is well mixed. These assumptions are conservative. The staff also acknowledges that as aerosols deposit, the bottom of a main steam pipe will become heated, relative to the top. This will induce a natural convection of gas within the pipe, trapping aerosols in the circulation zone and making it available to leak to the environment.

Keeping these uncertainties in consideration, the NRC staff has performed a sensitivity analysis to determine the effect of the potential overestimation of the aerosol removal, and found that the overall aerosol leakage through three MSIVs is greater than 5 percent, depositing less than 95 percent of the aerosol in the pipe. Given this information, the NRC staff concludes, based on its knowledge of aerosol physics and experience in previous AST license amendment reviews, that the Dresden and Quad Cities main steam line aerosol settling model which results in greater than 5 percent aerosol leakage to the environment is reasonable and appropriate.

Exelon also assumed deposition of elemental iodine in the main steam line piping. Exelon assumed 50 percent elemental iodine removal efficiency (decontamination factor of 2) in each steam line volume consistent with the value used in AEB-98-03, Appendix B. The NRC staff finds that the elemental iodine removal efficiency used by Exelon is conservative and, therefore, acceptable. Because elemental deposition is not gravity dependent, Exelon assumed elemental iodine deposition occurs on the entire surface area of the horizontal and vertical piping. Exelon has taken no exception or departure from the guidance provided in RG 1.183 for evaluating the radiological consequence resulting from this fission product release pathway.

Based on the above analysis, the NRC staff finds that Exelon’s modeling of the release by leakage through the MSIVs has been modeled conservatively and appropriately.

3.1.1.3 Post-LOCA Leakage From Engineered Safety Features Outside Containment

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 ECCS. Since portions of these systems are located outside of the primary containment, leakage from these systems is evaluated as a potential fission product release pathway. For the purpose of assessing the consequences of leakage from the ECCS, 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 assumed to be in both the primary containment atmosphere and in the suppression pool thus available for ECCS leakage concurrently. In a mechanistic treatment, the radioiodines in the primary containment atmosphere would relocate to the suppression pool over time. Noble gases released from the fuel are assumed to remain in the drywell atmosphere. Since aerosols and particulate radionuclides are not expected to become airborne on release from the ECCS, they are not included in the ECCS source term. These assumptions are consistent with the guidance provided in RG 1.183.

The analysis considers the equivalent of 2 gallons per minute (gpm) ECCS fluid leakage starting at the onset of the LOCA. This leakage rate includes a factor of 2 multiplier over the TS limit, in accordance with guidance provided in RG 1.183, to address increases in the leakage due to normal material degradation between surveillance tests. Exelon assumed 10 percent of the iodine in the ECCS leakage becomes airborne and is available for release. As was assumed for the primary containment leakage pathway, the fission products in the containment atmosphere are further mitigated by dilution in the secondary containment and by the SGT system filtration. Consistent with the guideline provided in RG 1.183, the licensee assumed leakage from the primary containment would mix with the reactor building free air with no more than 50 percent mixing efficiency. The leakage enters the environment via the SGT system as a filtered elevated release. Exelon has taken no exception or departure from the guidance provided in RG 1.183 for evaluating the radiological consequence resulting from this fission product release pathway.

3.1.1.4 Suppression Pool Post-LOCA pH Control

The regulatory guidance in RG 1.183 provides that the iodine released to the containment includes 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent iodine in organic forms. This iodine species assumption is only applicable if the suppression pool water is maintained at a pH of 7.0 or higher to ensure against elemental iodine re-evolution. Exelon proposes to use the standby liquid control system (SLCS) to inject sodium pentaborate to the RPV, where it will mix with ECCS flow and spill over to the drywell and then to the suppression pool. Sodium pentaborate, which is a base, will neutralize acids generated in the post-accident primary containment environment. The SLCS at Dresden and Quad Cities consists of two positive displacement pumps, each capable of injecting 40 gpm into the reactor vessel. Credit for the SLCS in the radiological analyses is based on operation of one SLC pump, manually initiated, and injection of the required amount of sodium pentaborate in the reactor and its transport to and mixing with the suppression pool water within 24 hours after initiation of the accident. The SLCS is manually initiated from the main CR. The licensee provided additional information regarding the SLCS and suppression pool pH control in its letters dated March 21 and 28, 2003, and June 30, August 6 and September 10, 2004.

3.1.1.4.1 pH Analysis

The NRC staff evaluated the licensee's proposed methodology for controlling the suppression pool pH after a LOCA, which is needed in order to meet 10 CFR 50.67. After a LOCA, several acidic species are introduced into the suppression pool. The main sources of acidic species are hydrochloric and nitric acids. Hydrochloric acid is generated by a decomposition of cable Hypalon and PVC insulation. Only the insulation on the cables exposed directly to radiation fields are decomposed. Nitric acid is produced by irradiation of water and air in the radiation environment existing in the containment after a LOCA. The only significant source of basic species is cesium hydroxide released from damaged fuel. With these chemical species and without the buffering action of sodium pentaborate, the pH in the suppression pool water will drop below seven in about one day. However, after adding a sufficient amount of buffer, the pH in the suppression pool could be maintained above seven for 30 days.

By letter dated March 28, 2003, the licensee responded to a request for additional information (RAI) by providing clarifying information on the methods being proposed in their request. The calculations, included in the licensee's response, demonstrated how the minimum amount of sodium pentaborate needed to produce the required suppression pool pH was determined. This analysis showed that upon the initiation of the injection, a minimum of 3769.4 pounds of sodium pentaborate will be delivered into the suppression pool within 24 hours following a

LBLOCA. The sodium pentaborate solution injected to the suppression pool will be supplied by the SLCS. Although the primary function of the SLCS is to introduce negative reactivity to the core in the event of a control rod mechanism failure, Exelon proposes to inject sodium pentaborate solution into the reactor for suppression pool pH control after a LOCA. The AST analysis specifies manual initiation of SLCS. In a letter dated August 6, 2004, the licensee provided the time sequence for SLCS initiation, and its transport and mixing with the suppression pool water. The time sequence provides for manual SLCS initiation and injection to start 2 hours after the beginning of the accident. Injection is completed when an adequate volume of sodium pentaborate solution is introduced into the suppression pool in about 4 ½ hours, well within the 24 hours assumed in the LOCA analyses. The licensee also provided the NRC staff with the necessary information needed to validate its claims. The NRC staff performed independent calculations to support its conclusion. The NRC staff evaluated the basis and input data for the different calculations made by the licensee, the chemical species that were being credited in the report and the implementation of the SLCS for the purposes described in their request. The licensee's results were found to be conservative and acceptable. The analysis done by the licensee was done only for Dresden, Unit 1 but the licensee properly demonstrated that this analysis bounded the analyses for the other Dresden unit and both units in Quad Cities. Also, there is a provision for the SLCS pumps, valves, and controls to be powered from diesel generators in the absence of normal power. The proposed TS change submitted by the licensee requires the SLCS to be maintained in an operable status whenever the reactor is in Mode 1 (run), 2 (start-up), or 3 (hot shutdown).

The licensee described the methodology for controlling the post-LOCA pH in the suppression pool water above seven. The methodology relies on using buffering action of sodium pentaborate, introduced into the suppression pool from the SLCS. The licensee provided analyses justifying that 3769.4 pounds of sodium pentaborate will ensure that pH in the suppression pool will stay above seven for 30 days after a LOCA. The licensee also determined that the SLCS could be activated when the plant is in Modes 1, 2, or 3. The NRC staff found the licensee's pH analyses in support of the application of an AST methodology to be acceptable. The acceptance was justified by the results of the NRC staff's review of the analysis provided by the licencee, thus concluding that the methodology presented in the licensee's submittal will ensure the suppression pool pH will stay basic for the period of 30 days after a LOCA.

3.1.1.4.2 Standby Liquid Control System

The proposed change extends the Applicability of TS 3.1.7 from Modes 1 and 2 to Modes 1, 2, and 3, and adds Required Action C.2. This action requires the plant be placed in Mode 4 (cold shutdown) in 36 hours if SLCS can not be restored within the required time. These changes implement the AST methodology regarding the use of SLCS to buffer the suppression pool following a LOCA involving fuel damage.

The NRC staff reviewed the SLCS with respect to its 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 SLCS is a safety-related role. The licensee stated that the SLCS is designated as a safety-related system.

The NRC staff reviewed the licensee's submittals and their response to RAIs on the use of the

SLCS for the safety-related function in a letter dated June 30, 2004. From the licensee's statements, the NRC staff has concluded the following:

> The SLC system is designated a safety-related system. As such, the SLCS, 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,
>
> a. the system components required for reactivity control and new suppression pool pH control functions are seismically qualified;
>
> b. the system is provided with emergency power with the capability to supply power from the emergency diesel generators;
>
> c. the system is subject to the American Society of Mechanical Engineers Code, Section XI, "Inservice Inspection Requirements," as required by 10 CFR 50.55a, "Codes and Standards;"
>
> d. the system is within the scope of the 10 CFR 50.65 Maintenance Rule;
>
> e. most components (pumps, squib valves, etc.) are redundant in parallel trains powered from different electrical busses; the exceptions are the containment isolation check valves and the selector switch in the main control room (these are discussed below under single failure review);
>
> f. emergency operating procedures (EOPs) direct the activation of the SLC following a LOCA when reactor water level can not 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 for pH control is injected; and
>
> g. 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 and the main CR selector switch. The containment isolation valves are one-and-a-half-inch, stainless steel valves mounted horizontally in the injection line. In the periodic inspections and testing of these valves, neither Quad Cities nor Dresden has experienced any failures of these valves to open on demand. A review of the industry database (EPIX) was performed and no failures of check valves of this type and manufacture failing to open were identified. Although acknowledging that a single failure to open, of one of the two check valves, could prevent SLC injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement, periodic testing and inspection, and historical performance of the component. The NRC staff finds that the use of a single penetration of the containment with the identified check valves as described by the licensee is acceptable.

The NRC staff also acknowledges that the selector switch in the main CR could fail and prevent either train, or both trains of injection, from functioning. The NRC staff determined that the switch was a highly reliable component at an accessible location. The switch could easily be replaced or bypassed to start one of the SLC trains if the switch were to fail. The NRC staff finds this to be acceptable.

The NRC 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 ensure core cooling after a LOCA. Procedure changes will be implemented to ensure that the SLCS is initiated within 2 hours of accident initiation when there is indication of fuel damage. In a letter dated September 10, 2004, the licensee stated that procedure changes will also be made to modify the suppression pool cooling return flow path and will ensure that LPCI flow is maintained through the vessel in order to ensure that the sodium pentaborate is transported from the reactor vessel to the suppression pool.

Using the LPCI in this mode for suppression pool cooling also provides mixing. The NRC staff concludes that there would be mixing and transport at some rate, and that it is 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 3.1.7 provide the capability of injecting SLC during hot shutdown. This would not be necessary for the anticipated transients without scram (ATWS) function of the SLC, but is needed 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 NRC staff finds these changes acceptable.

3.1.1.4.3 Human Factors Engineering

In its October 10, 2002, letter, the licensee indicated that, as a result of using AST methodology, the DBA LOCA analysis takes credit for minimizing the re-evolution of elemental iodine from the suppression pool, which is strongly dependent upon suppression pool pH. The licensee's analysis assumed that the borated solution (sodium pentaborate) was injected within 24 hours of the onset of a DBA LOCA and mixed within the suppression pool. The analysis demonstrated that the suppression pool pH 30 days after the LOCA, is greater than seven.

In response to the NRC staff's RAI, the licensee indicated in a letter dated March 21, 2003, that the analysis assumed that injection of a borated solution is manually initiated and that a minimum of 3769.4 lbs. of sodium pentaborate (or equivalent) is delivered into the suppression pool within 24 hours following a DBA LOCA. The current design function of the SLCS is to provide a backup method, independent of control rods, to make the reactor subcritical over a full range of operating conditions. This includes bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon-free state without taking credit for control rod insertion. Initiating the SLCS following fuel damage to control suppression pool pH is a new operator action during a DBA LOCA response that will be required after implementation of AST.

In addition to the design function, manual initiation of SLC is directed by the current Dresden and Quad Cities EOPs following a LOCA when reactor water level cannot be maintained above the top of active fuel. The licensee indicated that manual initiation of SLC also is directed by SAMGs which are entered when adequate core cooling cannot be maintained. The licensee described, for both Dresden and Quad Cities, the specific visual and auditory cues and various procedural directions that are present to operators when manual initiation of SLC is required. The licensee also stated that one current Dresden EOP, DEOP 500-3, “Alternate Water Injection Systems,” contains steps that may prevent the operator from injecting boron or meeting the assumed concentrations when manually initiating SLC. The licensee stated that prior to implementing the AST amendment, DEOP 500-3 will be revised to specifically direct boron injection without dilution until the required amount of boron is injected for proper pH control after a LOCA.

The licensee indicated that crediting manual initiation of SLC is appropriate since there is adequate time to initiate SLC under DBA LOCA conditions. The licensee’s analysis assumes that 3769.4 lbs. of sodium pentaborate (or equivalent) is delivered into the suppression pool within 24 hours following a DBA LOCA. This is equivalent to about 3000 gallons of sodium pentaborate solution (14 percent by weight). The SLCS at Dresden and Quad Cities consists of two positive displacement pumps, each capable of injecting 40 gpm into the reactor vessel. At 40 gpm, 75 minutes are needed to inject 3000 gallons of solution. There is sufficient time following a DBA LOCA (or any loss of adequate core cooling) to allow the required amount of sodium pentaborate to be injected within the 24-hour limit (refer to the time sequence of SLCS injection discussed in Section 3.1.1.4.1 above). Furthermore, initiating SLC is accomplished entirely from the main CR with a simple key-lock switch manipulation. Actuating the switch is the only action necessary to inject SLC into the reactor vessel. Operators in the CR have indication of proper SLC initiation which includes explosive valve continuity lights extinguished, flow light illuminated, reactor water clean-up system isolation, SLC tank level decrease, and adequate SLC pump discharge pressure.

At Quad Cities, since the cues and required actions for initiating SLCS are not changing for implementing AST, no changes to procedures and training are required. At Dresden, DEOP 500-3 will be revised as previously described and Dresden will train operators on the revised procedure.

The licensee also indicated in its March 21, 2003, letter that if operators failed to complete the required manual initiation of SLC as described, additional elemental iodine may re-evolve from the suppression pool from a DBA LOCA involving fuel damage resulting in increased CR and offsite doses. However, the licensee did not evaluate dose consequences of failing to complete the required manual action since manual initiation of the SLC system involves a single action that is addressed by procedures and operators are trained to accomplish. In addition, the licensee has determined that there is significant time margin for operators to manually initiate SLC before acceptable dose limits would be exceeded (see Section 3.1.1.4.1 above for a discussion of manual SLCS initiation time sequence). The licensee further stated in its March 21, 2003, submittal that there are no other new modified manual operator actions credited in the AST analysis.

The NRC staff concludes that the information described above provides reasonable assurance that the manual action proposed can be successfully performed without adverse safety consequences to the plant, plant personnel or the public.

3.1.1.5 Control Room Habitability

Exelon evaluated the dose to operators in the CR. Exelon assumed a CREVS activation time of 40 minutes following a LOCA and that the CR would not be isolated for that period. The CREVS has no CR air recirculation and draws in 2000 cubic feet per minute (cfm) of unfiltered outside air during normal mode of the CR HVAC system operation and 2000 cfm of filtered outside air during emergency mode of operation. The maximum measured unfiltered inleakage by a tracer gas testing during the emergency mode was 253 cfm (162 +/- 91 cfm) for Dresden and 297 cfm (222 +/- 75 cfm) for Quad Cities. Exelon assumed 400 cfm unfiltered inleakage during the emergency mode from 40 minutes to 30 days for the Dresden and Quad Cities radiological consequence analyses.

For the calculation of the CR operator’s dose, Exelon assumed that the normal CR ventilation system was operating for the first 40 minutes following a LOCA. For such an assumption to be valid, the normal CR ventilation system must have a source of emergency power. Dresden and Quad Cities do not have an emergency power source for the normal CR ventilation system.

In addition, the presumption of the operation of the normal CR ventilation system also necessitates that the inleakage characteristics of the control room envelope (CRE) be known. When Dresden and Quad Cities performed their tracer gas testing of the CRE to determine its inleakage characteristics, they did not determine the CRE’s inleakage characteristics with the normal CR ventilation system operating. In an RAI, the NRC staff indicated that Exelon needed to tracer gas test the CRE to determine the CRE’s inleakage characteristics while operating the normal CR ventilation system. In lieu of performing this test, Exelon performed a parametric study with CRE inleakage rates up to 62,200 cfm to demonstrate compliance with the dose acceptance criterion specified in GDC 19 and submitted the results of their parametric study with their submittal dated February 17, 2006. Although maximum makeup air into the CR during normal operation is 2200 cfm, Exelon used one CR air change per minute (approximately 60,000 cfm) for conservatism to allow for any unfiltered air inleakage.

Exelon’s parametric study showed a difference of less than 0.23 rem TEDE CR operator dose between 2000 and 62,200 cfm unfiltered air inleakage meeting the dose criterion specified in GDC 19 for an individual in the CR. The NRC staff confirmed Exelon’s results with its own confirmatory dose calculation. Therefore, the NRC staff finds that the CR habitability modeling to estimate exposure to an individual in the CR is acceptable and that the resulting doses are less than the GDC 19 criterion.

3.1.1.6 Radiological Consequence of Loss-of-Coolant Accident

Exelon evaluated the maximum 2-hour TEDE to an individual located at the EAB, the 30-day TEDE to an individual at the outer boundary of the LPZ, and the 30-day TEDE to an individual in the control room. The resulting doses are less than the 10 CFR 50.67 dose criteria. Based on its review discussed above, the NRC staff concludes that the licensee's application of the AST to the Exelon LOCA analysis is acceptable. The results of the licensee's radiological consequence calculation are provided in Tables 1 and 2. The major parameters and assumptions used by the licensee, and found acceptable by the NRC staff, are listed in Table 3.

3.1.2 Main Steamline Break

The MSLB accident considered is the complete severance of the largest main steam line inside the turbine building. The radiological consequences of a MSLB outside containment will bound the consequences of a break inside containment. Thus, only the MSLB outside of containment is considered with regard to the radiological consequences. The MSLB accident is described in the UFSAR Section 15.6.4, "Steam System Line Break Outside Containment." Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB.

The MSIVs are assumed to close within 5.5 seconds. This assumed time is based on the maximum allowed MSIV closure time in the TS (5.0 seconds for MSIV closure plus 0.5 seconds for instrument response). There is no core uncovery and, therefore, no fuel damage is projected for the design-basis MSLB. These assumptions are consistent with the current licensing basis as specified in the TS and described in the UFSAR, and are not affected by the AST implementation. Exelon used a mass coolant release of 1.4E+5 pounds during the MSLB accident in its dose calculation which is the maximum mass release value for all current BWR plants as provided in SRP Section 15.6.4 and bounds the current design-basis mass release value stated in the UFSAR Table 15.6-3.

The analysis is performed for two activity release cases, based on the maximum equilibrium and pre-accident iodine spike concentrations of 0.2 μCi/gm and 4.0 μCi/gm I-131 dose equivalent I-131, respectively. All of the accident activity was assumed to be released within 5.5 seconds following the accident as a ground level release, with no credit for turbine building holdup or dilution. These assumptions are in accordance with RG 1.183 guidance.

Exelon evaluated the maximum 2-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the dose acceptance criteria specified in RG 1.183 and SRP 15.0.1.

Exelon evaluated the dose to operators in the CR, using a puff-release CR atmospheric dispersion factor. The staff finds the use of the puff-release CR atmospheric dispersion factor acceptable because of the very short duration of the MSLB release (5.5 seconds). Exelon did not utilize atmospheric dispersion factors (χ/Q values) for the MSLB CR dose assessment. Instead, Exelon assumed that the CR would not be isolated during this event and no credit is taken for the operation of the CR emergency filtration system during this event. Inhalation and external exposure doses were determined based on radioactivity concentrations at the CR air intake. The resulting 30-day TEDE to an individual in the CR is less than the 10 CFR 50.67 dose criteria.

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 above and with those stated in the UFSAR as design bases. The results of the licensee's radiological consequence calculation are provided in Tables 1 and 2. The major parameters and assumptions used by the licensee, and found acceptable by the NRC staff, are listed in Table 5. Based on its review discussed above, the NRC staff concludes that the licensee's application of the AST to the Exelon MSLB accident analysis is acceptable.

3.1.3 Control Rod Drop Accident

This accident analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod. Localized damage to fuel cladding and a limited amount of fuel melt are projected. The fuel rod fission product inventory is based on long-term reactor operation at 102 percent of the rated thermal power and Exelon multiplied the fuel rod fission product inventory by a radial peaking factor of 1.7 to conservatively maximize the fission product release. A reactor trip will occur. Consistent with the current design basis, as described in the UFSAR Section 15.4.10.5, the main condenser mechanical vacuum pump is isolated on the main steam line radiation monitor high radiation signals following initiation of the CRDA. The MSIVs are assumed to remain open for the duration of the event. Exelon has projected that 850 fuel rods would be breached by the event, and of these damaged rods, 0.77 percent would exceed the threshold for melting. These projections are the current design basis and are not affected by the AST implementation.

Exelon analyzed two cases for radioactivity release from the CRDA. Case 1 (turbine/condenser leakage) assumes fission products will be transported through the main steam lines directly to the turbine and the main condenser. With tripping of the mechanical vacuum pumps by the main steam line radiation monitor (MSLRM) high radiation signals, the release will be from the turbine/condenser and gland sealing steam system. Consistent with the guidance provided in Appendix C to RG 1.183, Exelon assumed that 10 percent of the core inventory of noble gases and iodine is in the fuel gap and that 100 percent of the noble gases, 10 percent of the iodines and 1 percent of the alkali metals released reach the turbine and the main condenser due to plate-out in the RPV and main steam lines.

Of the iodine that enters the main condenser, 90 percent plates out. There is no reduction in noble gases. However, for the gland seal condenser, Exelon did not take any credit for iodine removal and conservatively assumed that 100 percent of iodine that reaches the gland seal condenser is available for release to the environment. The fission product gases in the main condenser are released at a rate of 1 percent by volume over 24 hours as a ground level release.

Case 2 (augmented off-gas system (AOG) release) assumes that the CRDA occurred during steam jet-air ejector (SJAE) operation. The AOG processes SJAE flow through the charcoal delay beds, which would eliminate iodine releases and delay noble gas releases allowing for decay. Releases from this leakage pathway are also assumed to be at ground level.

Exelon evaluated the maximum 2-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the dose acceptance criteria specified in RG 1.183 and SRP 15.0.1.

Exelon evaluated the dose to operators in the CR. It was assumed that the CR would not be isolated during the event and the normal CR ventilation system was operating for this event. In both cases analyzed for this event, the CREVS was not assumed to be operational. Exelon used 64,000 cfm unfiltered air inleakage into the CRE for Dresden (and 58,300 cfm for Quad Cities) for conservatism to allow for any unfiltered air inleakage as it did for the first 40 minutes for a LOCA (see Section 3.1.1.5 above). Exelon analyzed the CR dose over a 30-day period. The resulting 30-day TEDE to an individual in the CR is less than the 10 CFR 50.67 criteria. The NRC staff performed an independent confirmatory dose calculation (1) using 60,000 cfm unfiltered air inleakage and (2) using 2200 cfm normal makeup air flow with the CR operator being located at the CRE air intake. The NRC staff's dose calculations for both cases also meet the GDC 19 dose criterion.

The NRC staff finds that the licensee's CRDA analysis assumptions and methodology are consistent with the guidance of RG 1.183. The assumptions found acceptable to the NRC staff are presented in Table 4. The EAB, LPZ, and CR doses estimated by Exelon for the CRDA were found to be acceptable, and are listed in Tables 1 and 2. The NRC staff performed independent calculations and confirmed Exelon's results. Based on this review, the NRC staff concludes that the licensee's application of the AST to the Exelon CRDA analysis is acceptable.

3.1.4 Fuel-Handling Accident

This accident analysis postulates that a spent fuel assembly is dropped on top of the reactor core during refueling. Exelon assumed that the earliest at which the FHA could occur would be 24 hours following reactor shutdown as the movement of recently irradiated fuel will not occur earlier than this time period. A radial peaking factor of 1.7 was assumed for the damage rods. In its response dated September 3, 2005, to the NRC staff's RAI, Exelon stated that the current design and licensing bases for the postulated FHA is based on the failure of 111 fuel rods and 7X7 fuel rod array. However, Exelon further stated that there are no such fuel assemblies in operation in Dresden or Quad Cities at the present time. The majority of the current operating fuel is GE-14 10X10 array fuel with an equivalent pins per assembly of 87.33. With this fuel rod array configuration, 172 fuel rods are assumed to be damaged per NEDC-32868P GE14, "GESTAR Compliance Document," as referenced in Section 15.7 of the Dresden and Quad Cities UFSARs. Exelon stated that the fission product release source term would be higher with the failure of 111 fuel rods with 7X7 fuel array than with the GE-14 10X10 fuel array with 172 damaged fuel rods. The NRC staff finds that Exelon's analysis is conservative with respect to the assumption of fuel rod array in the release source term and is, therefore, acceptable. The fission product inventory in the fuel rod gap of the damaged 111 fuel rods is assumed to be instantaneously released because of the FHA.

The chemical form of the iodine released to the environment was determined by the chemical form of the iodine released from the gap and the decontamination factor (DF) associated with the water above the damaged fuel. The iodine in the gap was assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine and 0.15 percent organic iodine. The CsI was assumed to completely dissociate in the water. Since the minimum amount of water depth above the damaged fuel assembly is 19 feet at Dresden and Quad Cities, Exelon calculated a

DF of 135 for the elemental iodine in the pool water based on the elemental iodine DF of 500 (for 23 feet of pool water coverage) given in RG 1.183, Appendix B. The NRC staff finds that the calculated elemental iodine DF of 135 for 19 feet water coverage is acceptable.

Exelon assumed that the activity released from the water would be released to the reactor building and then to the environment via the reactor building vent stack. No credit was taken for reactor building holdup or dilution. The release from the reactor building vent was assumed to occur with zero vent velocity. Exelon assumed no credit for operating the SGTS nor for the CREVS. Exelon also assumed that all radioactivity released from the water was released within 2 hours.

Exelon evaluated the maximum 2-hour TEDE to an individual located at the EAB and the 30-day TEDE to an individual at the outer boundary of the LPZ. The resulting doses are less than the dose acceptance criteria specified in RG 1.183 and SRP 15.01. The NRC staff has performed independent dose calculations of the offsite and onsite consequences of an FHA. Table 6 contains details of the assumptions utilized by Exelon and acceptable to the NRC staff. The results of Exelon's calculations are presented in Tables 1 and 2. Both the onsite and offsite doses were found to be acceptable for the proposed LAR.

Exelon evaluated the dose to operators in the CR. Exelon assumed that (1) the CR would not be isolated during the event, (2) the CREVS was not assumed to be operational, and (3) the normal CR ventilation system was operating for this event. Exelon used 6.4E+4 cfm unfiltered air inleakage into the CRE for Dresden (and 5.83E+4 cfm for Quad Cities) for conservatism to allow for any unfiltered air inleakage as it did for the first 40 minutes for a LOCA (see Section 3.1.1.5 above). Exelon analyzed the CR dose over a 30-day period. The resulting 30-day TEDE to an individual in the CR is less than the 10 CFR 50.67 criteria. The NRC staff performed independent confirmatory dose calculations (1) using 60,000 cfm unfiltered air inleakage and (2) using 2200 cfm normal makeup air and the CR operator located at the CRE air intake. The NRC staff's dose calculations for both cases met the requirements of 10 CFR 50.67 and the GDC 19 dose criteria. Based on this review, the NRC staff concludes that the licensee's application of the AST to the Exelon FHA analysis is acceptable.

3.2 Atmospheric Dispersion Estimates

Exelon used onsite meteorological data collected during calendar years 1995-1999 to generate new CR, EAB, and LPZ atmospheric dispersion factors (χ/Q values) for the Dresden and Quad Cities sites. The licensee used the ARCON96 and PAVAN computer codes to generate χ/Q values for the LOCA, CRDA, and FHA. The licensee did not utilize χ/Q values for the MSLB CR dose assessment (see Section 3.1.2 above). Exelon calculated EAB and LPZ χ/Q values for the MSLB using the methodology and assumptions described in RG 1.5. The resulting χ/Q values represent a change from the χ/Q values used in the current Dresden and Quad Cities UFSAR, Chapter 15, "Accident Analyses."

3.2.1 Meteorological Data

Exelon generated new χ/Q values for the Dresden and Quad Cities LOCA, CRDA, and FHA dose assessments using site meteorological data collected during 1995-1999 at both sites. These data were originally provided for NRC staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code) and joint wind speed, wind direction, and atmospheric stability frequency distributions (for input to the PAVAN atmospheric dispersion computer code). The data were provided in a letter to the NRC staff dated August 4, 2003. Following an initial review of the data, the NRC staff requested additional information concerning the format of the data. In its November 5, 2004, response, Exelon provided corrected meteorological data files. The NRC staff performed a quality review of the 1995-1999 corrected hourly meteorological databases, using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data."

Wind speed and wind direction data were measured on the Dresden onsite meteorological tower at heights of 10.7 meters, 45.7 meters, and 91.4 meters above the ground. Temperature difference data, which were used to determine atmospheric stability class, were measured between the 45.7-meter and 10.7-meter levels and between the 91.4-meter and 10.7-meter levels. Wind speed and wind direction data were measured on the Quad Cities onsite meteorological tower at heights of 10.1 meters, 59.7 meters, and 90.2 meters above the ground. Temperature difference data were measured between the 59.7-meter and 10.1-meter levels and between the 90.2-meter and 10.1-meter levels.

The combined data recovery of the wind speed, wind direction, and stability (temperature difference) data was in the upper 90 percentiles at each level during each of the 5 years at both the Dresden and Quad Cities sites. With respect to atmospheric stability measurements, the time of occurrence and duration of stable and unstable conditions were consistent with expected meteorological conditions. Stable and neutral conditions were reported to occur at night and unstable and neutral conditions during the day, with neutral or near-neutral conditions predominating during each year. Wind speed, wind direction, and stability class frequency distributions for each measurement channel were reasonably similar from year to year and among all three levels. A comparison of joint frequency distributions derived by the NRC staff from the ARCON96 hourly data with the joint frequency distributions developed by Exelon for input into PAVAN showed reasonable agreement.

Exelon stated that during 1995-1999, the onsite meteorological measurement programs at both the Dresden and Quad Cities sites met the guidance of RG 1.23. Calibrations of the measurement systems were performed onsite every other month until mid-1997, then on a quarterly basis after mid-1997. Data were accessed remotely on a daily basis and checked initially by computer with subsequent review by an experienced meteorologist. The data were checked for consistency with other tower measurements and with respect to local conditions. Further review was performed by additional computer checks and comparison of data with local National Weather Service reported data. Suspected problems were reported for measurement systems checks, typically within 2 days of identification.

For the reasons cited above, the NRC staff has concluded that the 1995-1999 meteorological data measured at the Dresden and Quad Cities sites provide an acceptable basis for making atmospheric dispersion estimates for use in the LOCA, CRDA and FHA dose assessments performed in support of the LAR.

3.2.2 Control Room Atmospheric Dispersion Factors

Exelon generated new CR χ/Q values for postulated ground level (i.e., MSIV and reactor building vent) and elevated (i.e., station chimney) releases from Dresden and Quad Cities for LOCA, CRDA and FHA using guidance provided in RG 1.194. These new CR χ/Q values were calculated using the ARCON96 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 CR χ/Q values for use in DBA radiological analyses. The PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations") was also used as discussed in RG 1.194 in the calculation of CR χ/Q values for postulated elevated releases from the station chimney. All releases were modeled as point releases.

For postulated ground level releases at Dresden, Exelon executed ARCON96 using the 1995-1999 onsite hourly 10.7-meter and 45.7-meter wind data and 45.7-meter to 10.7-meter temperature difference data described above. For postulated ground level releases at Quad Cities, Exelon executed ARCON96 using the 1995-1999 onsite hourly 10.1-meter and 59.7-meter wind data and 59.7-meter to 10.1-meter temperature difference data described above.

Releases from the MSIVs were assumed to be at ground level but would actually be below grade. Since these release points are less than 2½ times the height of adjacent structures, they were modeled with ARCON96 as point sources using the ground-level release mode option in conformance with RG 1.194 guidance. Exelon generated values for both units and selected the limiting χ/Q values at each site for use in the dose assessments. The resulting CR CRDA χ/Q values are listed in Table 8 and Table 12 for the Dresden and Quad Cities sites, respectively.

The reactor building vent exhaust stack is at a height of 48.6 meters above the ground at both the Dresden and Quad Cities sites. Since this release point is less than 2½ times the height of adjacent structures, Exelon modeled the FHA with ARCON96 as a point source using the ground-level release mode option in conformance with RG 1.194 guidance. The resulting CR FHA χ/Q values are listed in Table 9 and Table 13 for the Dresden and Quad Cities sites, respectively.

Exelon generated new CR χ/Q values for postulated releases from the station chimney at both Dresden and Quad Cities to the respective site CR air intakes implementing guidance in RG 1.194 for elevated releases. RG 1.194 states that χ/Q values for postulated elevated releases to the CR should be based on χ/Q values generated from both the ARCON96 and PAVAN computer codes. However, Exelon used the more conservative PAVAN generated χ/Q values only for all time periods in the CR dose assessments for postulated releases from the station chimneys.

For elevated releases, RG 1.194 states that χ/Q values for several distances in each wind direction should be generated using PAVAN with the objective of identifying the maximum χ/Q value. To address this goal, the licensee calculated new PAVAN χ/Q values at approximately 20 distances between 75 and 3000 meters to estimate the limiting case. When generating the PAVAN χ/Q values for the Dresden site, the licensee input an effective stack height of 83.3 meters, which is the vertical distance between the 94.6-meter stack and the CR intake at a height of 11.3 meters. When generating the PAVAN χ/Q values for the Quad Cities site, the licensee input an effective stack height of 85.8 meters, which is the vertical distance between the 94.6-meter stack and the CR intake at a height of 8.8 meters.

The meteorological input for the PAVAN χ/Q values consisted of 1995-1999 joint frequency distributions compiled from wind data measured at the 91.4-meter level on the onsite meteorological tower at Dresden and at the 90.2-meter level at Quad Cities. For Dresden, the meteorological input to the ARCON96 χ/Q values consisted of 1995-1999 hourly wind speed and direction data from the 10.7-meter and 91.4-meter levels on the onsite meteorological tower. Stability class input to both the PAVAN and ARCON96 χ/Q values was based on temperature difference measurements made between the 91.4-meter and 10.7-meter levels on the onsite meteorological tower. For Quad Cities, the meteorological input to the ARCON96 χ/Q values consisted of 1995-1999 hourly wind speed and direction data from the 10.1-meter and 90.2-meter levels on the onsite meteorological tower. Stability class input to both the PAVAN and ARCON96 χ/Q values was based on temperature difference measurements made between the 90.2-meter and 10.1-meter levels on the onsite meteorological tower. The resulting ARCON96 and PAVAN χ/Q values are then combined as a function of time period to determine the effective χ/Q values. The resultant effective χ/Q values were less than the χ/Q values generated using PAVAN only. As a result, Exelon used the more conservative PAVAN generated χ/Q values for all time periods in the CR dose assessments for releases from the station chimneys.

In summary, the NRC staff evaluated the applicability of the ARCON96 and PAVAN models and concludes that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of these models in support of this LAR for the Dresden and Quad Cities sites. The NRC staff qualitatively reviewed the inputs to the ARCON96 and PAVAN calculations and finds them acceptable when compared with site configuration drawings and NRC staff practice. The NRC staff made an independent evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 and PAVAN computer model for several random cases and obtained similar results. On the basis of this review, the NRC staff concludes that the χ/Q values for the Dresden and Quad Cities LOCA, CRDA, and FHA releases to the CR as presented in Tables 7 through 10 for the Dresden site and Tables 11 through 14 for Quad Cities are acceptable for use in the DBA CR dose assessment performed in support of this LAR.

3.2.3 Offsite Atmospheric Dispersion Factors

Exelon generated ground-level release LOCA, CRDA, and FHA and elevated release LOCA χ/Q values for the EAB and LPZ using the methodology described in RG 1.145, and the PAVAN computer code. For the Dresden ground-level release analysis, Exelon input the joint frequency distribution derived from the 1995-1999 onsite 10.7-meter wind data. Stability class was calculated using the temperature difference between the 45.7-meter and 10.7-meter levels on the meteorological tower. Elevated release χ/Q values were generated from the joint frequency distribution derived from the 91.4-meter wind data with stability class calculated using the temperature difference between the 91.4-meter and 10.7-meter levels. With regard to the

Quad Cities ground-level release analysis, Exelon input the joint frequency distribution derived from the 1995-1999 onsite 10.1-meter wind data. Stability class was calculated using the temperature difference between the 59.7-meter and 10.1-meter levels on the meteorological tower. Elevated release χ/Q values were generated from the joint frequency distribution derived from the 90.2-meter wind data with stability class calculated using the temperature difference between the 90.2-meter and 10.1-meter levels.

The NRC staff qualitatively reviewed the inputs to the PAVAN computer calculations and finds them acceptable when compared with site configuration drawings and NRC guidance. For postulated ground-level releases, Exelon assumed a building minimum cross-sectional area of 1545 and 1564 square meters at Dresden and Quad Cities, respectively. The licensee input EAB and LPZ distances of 800 meters and 8000 meters, respectively, in calculations for the Dresden site and EAB and LPZ distances of 380 meters and 4828 meters, respectively, in calculations for Quad Cities. Exelon used the elevated-release mode option of the PAVAN computer code to model the release from the free-standing station chimneys at the Dresden and Quad Cities sites, assuming fumigation during the 0-0.5 hour time period as specified in RG 1.145 for inland sites. As stated before, the station chimney at each site is 94.6 meters tall.

Exelon modeled the reduction in effective stack height due to topography by inputting actual terrain heights for the EAB. For the LPZ, Exelon calculated the bounding case by inputting the height of the top of the station chimney as the terrain height in each direction, thus reducing the effective release height to ground level. As a result of this conservative assumption, as shown in Table 7 and Table 11 for the Dresden and Quad Cities sites, respectively, the calculated 0.5-2 hour LPZ elevated release χ/Q value is higher than the 0.5-2 hour elevated release EAB and 0-2-hour CR χ/Q values. The NRC staff performed an independent evaluation of the resulting atmospheric dispersion estimates for the postulated ground-level and elevated releases using PAVAN and finds the Exelon calculations acceptable.

Exelon calculated EAB and LPZ atmospheric dispersion factors for the MSLB event at both Dresden and Quad Cities 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 at Dresden are presented in Table 10 and in Table 14 for the Quad Cities site.

In summary, the NRC staff reviewed Exelon's assessment of the Dresden and Quad Cities EAB and LPZ post-accident dispersion conditions generated from Exelon's meteorological data and atmospheric dispersion modeling and finds them to be acceptable for the application in which they are being used. On the basis of this review, the NRC staff concludes that the resulting Dresden EAB and LPZ χ/Q values presented in Tables 7 through 10 and Quad Cities EAB and LPZ χ/Q values presented in Tables 11 through 14 are acceptable for use in the LOCA, CRDA, FHA and MSLB accident dose assessments performed in support of this LAR.

3.3 <u>Technical Specification Changes</u>

The following sections describe the changes the licensee proposed to the Dresden and Quad Cities TS. Some of the proposed TS changes are justified under the guidance of the NRC-approved TSTF-51. The remainder of the requested changes address the AST methodology, which affects the definitions, the SLCS, the ventilation filter test program requirements (later withdrawn), containment leakage test requirements, and MSIV leakage SRs.

3.3.1 TSTF-51 Changes

Proposed TS changes associated with the implementation of the TSTF-51 traveler provides for relaxation of certain requirements during core alteration and movement of irradiated fuel. The purpose of the TSTF-51 TS changes is to establish a point where OPERABILITY of ESFs typically used to mitigate the consequences of a FHA are no longer required to meet the SRP guidance on offsite dose limits (i.e., less than 25 percent of the 10 CFR Part 100, "Reactor Site Criteria," limits or the limits specified in 10 CFR 50.67). Specifically, the proposal identifies that only "recently" irradiated fuel contains sufficient fission products to require OPERABILITY of the accident mitigation features to meet the accident analysis assumptions. Therefore, the APPLICABILITY requirements for the associated mitigation features (including the electrical support systems) are revised. The requested changes would eliminate TS requirements for ESFs during core alterations by deleting "During CORE ALTERATIONS" from APPLICABILITY. The requested change also adds "recently" to "irradiated fuel" to revise APPLICABILITY to "During movement of recently irradiated fuel." The affected TS Limiting Conditions for Operation (LCO) required ACTION statement to immediately suspend movement of irradiated fuel assemblies in secondary containment, when the LCO is not met, is also revised to require such action only when recently irradiated fuel assemblies are moved. The affected TS are defined in the following Table. These changes apply to both units at Dresden and Quad Cities except where noted.

TS Section Number	Title	Extent of change
Table 3.3.6.2-1, Note b	Secondary Containment Isolation Instrumentation	deletes reference to core alterations. adds "recently" in front of "irradiated fuel"
Table 3.3.7.1-1, Note b (Quad Cities only)	CREV System Isolation Instrumentation	deletes reference to core alterations. adds "recently" in front of "irradiated fuel
3.3.7.1 Applicability (Dresden only)	CREV System Isolation Instrumentation	deletes reference to core alterations. adds "recently" in front of "irradiated fuel"
3.6.4.2 Applicability and Action D3.8.2	SCIVs	deletes reference to core alterations. adds "recently" in front of "irradiated fuel"
3.6.4.3 Applicability and Actions C and F	SGT System	deletes reference to core alterations. adds "recently" in front of "irradiated fuel"

TS Section Number	Title	Extent of change
3.7.5 Applicability and Action C	Control Room Emergency Ventilation AC System	deletes reference to core alterations. adds “recently” in front of “irradiated fuel”
SR 3.8.1.21	AC Sources - Operating	adds “recently” in front of “irradiated fuel”
3.8.2 Applicability and Actions A and B	AC Sources - Shutdown	adds “recently” in front of “irradiated fuel”
3.8.5 Applicability and Action A	DC Sources - Shutdown	adds “recently” in front of “irradiated fuel”
3.8.8 Applicability and Action A	Distribution Systems - Shutdown	adds “recently” in front of “irradiated fuel”

In order to implement the above changes to APPLICABILITY and ACTION statements, the licensee proposed a revised FHA dose analysis for Quad Cities and Dresden that takes credit for a radioactive decay period of 24 hours, based on an AST pursuant to 10 CFR 50.67 and the guidance of RG 1.183 (Section 3.1.4 above). Given this decay period, the licensee is now proposing changes to redefine the TS requirements by making those ESF systems originally relied upon to mitigate an FHA applicable only for the movement of fuel that has been “recently irradiated.” The term “recently irradiated” represents the decay period for the reduction in radionuclide inventory available for release in the event of an FHA. As discussed in Section 3.1.4 above, the licensee stated that the earliest time an FHA could occur is 24 hours following reactor shutdown. As the FHA analyses considered an assumption of 24 hours decay period and conservative fuel parameters, a minimum of a 24-hour decay time to define “recently” is adequate. In summary, the licensee has demonstrated that once the reactor has been shut down for 24 hours, the results of the FHA re-analysis (that does not rely on either the secondary containment integrity, SGTS, or the CR isolation and emergency ventilation system) will not exceed offsite dose limitations.

In addition, consistent with the instructions in TSTF-51, Revision 2, regarding decreasing doses even further below those provided by natural decay, in a letter dated August 22, 2005, the licensee has committed to follow the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6, "Assessment Methods for Shutdown Conditions,” Subsection 5, "Containment - Primary (PWR)/Secondary (BWR).” The licensee stated that upon implementation of the approved AST license amendments, the following guideline will be included in the assessment of systems removed from service during movement of irradiated fuel:

> During fuel handling/core alternations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioacitivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay.

> A single normal or contingency method to promptly close primary or secondary containment penetration should also be developed. Such prompt methods need not completely block the penetrations or be capable of resisting pressure. The purpose of the "prompt methods" mentioned above is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

In discussing its commitment to NUMARC 93-01, in a letter dated September 3, 2004, the licensee stated that, since each station is a two unit facility that shares a common secondary containment, the secondary containment requirements can not be relaxed for a single unit. Specific procedures control open penetrations which require a person stationed at the penetration with sealing materials in communication with the CR. The secondary containment at each station is maintained at a negative pressure by either the reactor building ventilation system or the SGT system. These procedures also require monitoring of reactor building negative pressure. The licensee stated if the differential pressure is less than the required amount or if the SGT system initiates, then the penetration must be sealed. In its September 15, 2003, letter, the licensee also pointed out that refueling operation is performed with the reactor building ventilation system running, and if the radioactivity level exceeds the setpoint of the reactor building exhaust or the refuel floor radiation monitors, secondary containment is isolated and SGT system is initiated. This would reduce any release of contaminants to the public or CR operators and, thus, reduce dose. Although the FHA was analyzed with the reactor exhaust system operating (all activity is released directly to the environment through the reactor building vent stack, i.e., ground level release, over a 2-hour period without filtration or holding), isolation of the secondary containment promotes public health and safety by containing the release from an accident until it can be treated and monitored as appropriate.

The deletion of the term "CORE ALTERATIONS" is justified since an FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release, and such FHAs will be fully enveloped by the proposed APPLICABILITY and ACTION statements.

The proposed changes do not impact TS requirements for systems needed for decay heat removal, or requirements to maintain the specified water levels over irradiated fuel. Since the proposed revisions to the TS follow the guidance of the NRC-approved TSTF-51, the NRC staff concludes that these revisions are acceptable.

3.3.2 TS Section 3.1.7, "Standby Liquid Control System"

The proposed changes to the TS and the NRC staff safety evaluation is discussed in Section 3.1.1.4.2 above.

The extension of applicability to Mode 3 provides the capability of injecting SLC during hot shutdown. This would not be necessary for the 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 NRC staff finds these changes acceptable.

3.3.3 TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation"

The proposed changes extend the operability requirement for reactor water cleanup (RWCU) system primary containment isolation instrumentation upon SLCS initiation from Modes 1 and 2 to Modes 1, 2, and 3. As discussed in Section 3.1.1.4 above, the licensee proposes to use the SLCS to maintain suppression pool pH post LOCA to prevent re-evolution of iodine. The NRC staff's SE is discussed in Sections 3.1.1.4.1 thru 3.1.1.4.3. Isolation of RWCU is needed to prevent dilution of sodium pentaborate injected into the reactor by the SLCS. This proposed change is needed to implement the AST methodology. Therefore, the NRC staff finds that this change is acceptable.

3.3.4 TS Section 5.5.7 c and e, "Ventilation Filter Test Program," and Standby Gas Treatment Surveillance SR 3.6.4.3.1 (Withdrawn)

In its October 10, 2002, LAR, the licensee proposed changes to TS 5.5.7 paragraphs c and e, as follows:

a. increase the methyl iodide penetration for the SGT system filter testing from 2.5 percent to 50 percent;
b. increased the methyl iodide penetration for CREVS from 0.5 percent to 5 percent;
c. increase the relative humidity for SGT system filter testing from 70 percent to 95 percent (Quad Cities only); and
c. delete the SGT system heater test (Quad Cities only).

Additionally, the change proposed in SR 3.6.4.3.1 would delete "with heaters operating" from the requirement (Quad Cities only).

In a letter dated September 15, 2003, the licensee withdrew the above requested changes. Therefore, the NRC staff did not consider the above changes.

3.3.5 TS Definitions Section 1.1, "Dose Equivalent I-131"

The intent of the TSs on specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. The licensee currently calculates the dose equivalent I-131 using thyroid dose conversion factors, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole-body dose and thyroid dose as done previously. Therefore, the licensee proposed to use the inhalation committed dose conversion factors from Federal Guidance Report No. 11 (FGR No. 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submission, and Ingestion," and delete reference to TID-14844. The NRC staff finds the proposed use of FGR No.11 to be acceptable.

3.3.6 TS Section 5.5.12c, "Primary Containment Leakage Rate Testing Program"

The current design-basis maximum allowable containment leak rates (L_a) in the Dresden and Quad Cities TSs are 1.0 and 1.6 percent by weight of the containment atmosphere air mass per day at a design-basis LOCA maximum peak containment pressure of 48 psig for Dresden and Quad Cities, respectively. In this LAR, Exelon requested to increase the maximum allowable containment leak rate to 3 percent per day for the entire duration of the postulated LOCA (30 days) for both Dresden and Quad Cities. Contrary to RG 1.183, the licensee conservatively elected not to claim a reduction of the maximum allowable containment leak rate after 24 hours.

Based on the SE provided in Section 3.1.1.1 above, the NRC staff finds that these requested changes are acceptable from a radiological dose perspective.

3.3.7 TS Section SR 3.6.1.3.10, “MSIV Leakage Surveillance”

Exelon requested to increase the maximum allowable MSIV leak rate in the SR to read:

> Verify the leakage rate through each MSIV leakage path is # 34 scfh when tested at $ 25 psig, and the combined leakage rate for all leakage paths is # 86 scfh when tested at $ 25 psig.

The current SR states, “Verify the combined leakage rate for all leakage paths is equal or less than 46 scfh when tested at equal or greater than 25 psig.”

As discussed in Section 3.1.1.2 above, the licensee assumed a leakage rate for MSIVs equal to 150 scfh for all four main steam lines, and 60 scfh maximum through any one steam line in the accident analyses that maximized the effect of MSIV leakage. These leakage values in the LAR are based on a pressure of 48 psig. Exelon used a conversion factor of 1.73 to determine the corresponding leakage rates at the test pressure $ 25 psig.

Based on discussion in Section 3.1.1.2 above, the NRC staff finds that these requested changes are acceptable from a radiological dose perspective.

3.3.8 Conclusion on TS Changes

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Exelon to assess the radiological consequences of the proposed full implementation of an AST and the TS changes requested. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee’s estimates of the EAB, LPZ, and CR doses will comply with the criteria. The NRC staff finds, with reasonable assurance, that the Dresden and Quad Cities TS, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the NRC staff finds that the proposed TS changes to this LAR are acceptable.

This licensing action is considered to be a full implementation of an AST. With this approval, the previous accident source term in the Dresden and Quad Cities design basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole-body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Section 50.67 or fractions thereof, as defined in SRP 15.0.1. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the Dresden and Quad Cities design basis and modified by the present amendment.

3.4 TS Section 2.1.2, "Environmental Qualification"

The licensee has elected for Dresden and Quad Cities to retain the TID-14844 assumptions for performing the required environmental qualification (EQ) analyses. The radiation doses used for the EQ analyses at the current licensed core power level following extended power uprate (EPU) for 2957 MWt were further adjusted by 102 percent (3016 MWt) in support of AST evaluations.

The equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses. For the equipment exposed to sump water post-LOCA, the integrated doses calculated with the AST exceeded those calculated with TID-14844 after 145 days for a BWR, because of the 30 percent versus 1 percent release of cesium, according to NUREG-1465. The licensee's EQ program for Dresden and Quad Cities is based on a post accident period of 30 days. The continued use of the TID-14844 source term provides integrated doses for equipment that envelop those that would be calculated using AST for the duration of the EQ program consideration. Therefore, following implementation of AST, Dresden and Quad Cities will continue to meet their commitment to 10 CFR 50.49 by using a radiation environment associated with the most severe DBA.

The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety, and that operating reactors licensed under this approach would not be required to reanalyze accidents using the revised source term.

Based on the above information, the NRC staff concurs with the licensee's position to retain the TID-14844 assumptions for performing the required EQ analyses.

3.5 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Exelon to assess the radiological consequences of the proposed full implementation of an AST and requested TS changes. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff finds, with reasonable assurance that, Dresden and Quad Cities, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the NRC staff finds that the proposed LAR, is acceptable.

This licensing action is considered to be a full implementation of an AST. With this approval, the previous accident source term in the Dresden and Quad Cities design basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole-body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Section 50.67 or fractions thereof, as defined in SRP 15.0.1. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the Dresden and Quad Cities design basis and modified by the present amendment.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 49816; August 19, 2003). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

5. Letter from Keith R. Jury (Exelon) to U.S. NRC, “Request for License Amendments Related to Application of Alternative Source Term,” dated October 10, 2002, Agencywide Documents Access and Management System (ADAMS) Accession Number ML022940292.

6. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated March 21, 2003, ADAMS Accession Number ML030930030.

7. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated March 28, 2003, ADAMS Accession Number ML030990563.

8. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated August 4, 2003, ADAMS Accession Number ML032260099.

9. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated September 15, 2003, ADAMS Accession Number ML032671358.

10. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated October 31, 2003, ADAMS Accession Number ML033210069.

11. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated June 30, 2004, ADAMS Accession Number ML041830426.

12. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated August 6, 2004, ADAMS Accession Number ML042300472.

13. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated September 3, 2004, ADAMS Accession Number ML042580116.

14. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated September 10, 2004, ADAMS Accession Number ML042660141.

15. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated September 22, 2004, ADAMS Accession Number ML042730386.

16. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated November 2, 2004, ADAMS Accession Number ML0443150244.

17. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated November 5, 2004, ADAMS Accession Number ML043220162.

18. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated March 3, 2005, ADAMS Accession Number ML050630530.

19. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term," dated August 22, 2005, ADAMS Accession Number ML052430273.

20. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated September 3, 2005, ADAMS Accession Number ML052550370.

21. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated September 27, 2005, ADAMS Accession Number ML052780513.

22. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated February 17, 2006, ADAMS Accession Number ML060580573.

23. Letter from Patrick R. Simpson (Exelon) to U.S. NRC, “Additional Information Supporting the Request for License Amendments Related to Application of Alternative Source Term,” dated May 25, 2006, ADAMS Accession Number ML061460305.

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Table 1
Radiological Consequences Expressed as TEDE [1]
(rem)
Dresden Units 2 and 3

Design-basis accident	EAB [2]	LPZ [3]	Control Room
LOCA	1.58	8.68E-1	4.87
Dose criteria	25	25	5.0
Main steamline break accident[4]	8.48E-2	1.06E-2	1.89E-1
Dose criteria	2.5	2.5	5.0
Main steamline break accident [5]	1.70	2.12E-1	3.77
Dose criteria	25	25	5.0
Control rod drop accident [6]	5.55E-1	5.82E-2	1.06
Dose criteria	6.3	6.3	5.0
Control rod drop accident [7]	3.95E-1	6.61E-2	4.01
Dose criteria	6.3	6.3	5.0
Fuel Handling accident [8]	9.67E-1	1.01E-1	1.99
Dose criteria	6.3	6.3	5.0

(1) Total effective dose equivalent
(2) Exclusion area boundary
(3) Low population zone
(4) Equilibrium iodine coolant concentration (0.2 micro curies per gram)
(5) Maximum iodine coolant concentration (4.0 micro curies per gram)
(6) Gland seal condenser leakage and steam jet-air ejector release
(7) Main and gland seal condenser releases
(8) 19 feet water coverage in the pool

Table 2
Radiological Consequences Expressed as TEDE [1]
(rem)
Quad Cities Units 1 and 2

Design Basis Accidents	EAB [2]	LPZ [3]	Control Room
LOCA	8.47	2.63	4.08
Dose criteria	25	25	5.0
Main steamline break accident [4]	1.67E-1	1.68E-2	1.89E-1
Dose criteria	2.5	2.5	5.0
Main steamline break accident [5]	3.32	3.35E-1	3.79
Dose criteria	25	25	5.0
Control rod drop accident [6]	3.01	2.30E-1	8.33E-1
Dose criteria	6.3	6.3	5.0
Control rod drop accident [7]	2.14	2.57E-1	3.12
Dose criteria	6.3	6.3	5.0
Fuel Handling accident [8]	5.24	4.01E-1	1.80
Dose criteria	6.3	6.3	5.0

(1) Total effective dose equivalent
(2) Exclusion area boundary
(3) Low population zone
(4) Equilibrium iodine coolant concentration (0.2 micro curies per gram)
(5) Maximum iodine coolant concentration (4.0 micro curies per gram)
(6) Gland seal condenser leakage and steam jet-air ejector release
(7) Main and gland seal condenser releases
(8) 19 feet water coverage in the pool

Table 3
Parameters and Assumptions Used in Radiological Consequence Calculations for Loss-of-Coolant Accident

Parameter	Value
Reactor power	3016 MWt
Drywell air volume	1.58E+5 ft^3
Wetwell volume	
Dresden	1.20E+5 ft^3
Quad Cities	1.11E+5 ft^3
Reactor building free air volume	
Dresden	4.50E+6 ft^3
Quad Cities	4.70E+6 ft^3
Fraction of reactor building available for mixing	0.5
SGTS exhaust rate	4.0E+3 cfm (+/-10%)
SGTS filter efficiencies	
Elemental iodine	50%
Organic iodine	50%
Particulate aerosol	99%
Containment leak rates	
0 to 720 hour	3.0% per day
MSIV leak rates (scfh)	
Total for all four lines	150
One line with MSIV failed	60
First intact line	60
Second intact line	30
Third intact line	0
Aerosol removal rate constant	
0 to 720 hours	8.259 to 8.260 per hour
Elemental iodine removal efficiency in MSIV lines	50%
Suppression pool water volume	1.1E+5 ft^3
ECCS leak rates	
0 to 720 hours	2 gpm
Iodine partition factor	10%
Release points	
Containment and ESF leakages	station chimney
MSIV leakage	ground-level release via MSIV release pathway
Atmospheric dispersion factors	Tables 7 and 11

Table 3 (cont'd)
Parameters and Assumptions Used in
Radiological Consequence Calculations
for
Loss-of-Coolant Accident

Parameter	Value
CR volume ft^3	
Dresden	8.1E+4
Quad Cities	1.84E+5
CREV filtration system activation time	40 minutes
CREV system air intake rate	2000 cfm (+/-10%)
CR air recirculation rate	0
CR unfiltered air inleakage rate during emergency mode	
0 to 40 minutes	6.22E+4 cfm
40 minutes to 720 hours	400 cfm
CR atmospheric dispersion factors	Tables 7 and 11

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
for
Control Rod Drop Accident

Parameter	Value
Reactor power	3016 MWt
Reactor radial peaking factor	1.7
Fraction of fuel melted	0.0077
Number of fuel rods in core assumed to fail	850
Fraction of activity released from melted fuel	per RG 1.183
Fraction of fission product in fuel gap	per RG 1.183
Condenser vapor space volume	
Dresden	5.5E+4 ft^3
Quad Cities	9.2E+4 ft^3
Condenser leak rate	1.0% per day
AOG charcoal delay times	
Xenon	14.6 days
Krypton	19.4 hours
Release period	24 hours
Dose conversion factors	FGR 11 and FGR 12
Atmospheric dispersion factors	Tables 8 and 12
Control Room	
CREV system initiated	No
Normal ventilation system flow rate	2000 cfm (+/-10%)
Total assumed intake during normal operation	
Dresden	6.40E+4 cfm
Quad Cities	5.83E+4 cfm
Atmospheric dispersion factors	Tables 8 and 12

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
for
Main Steamline Break Accident

Parameter	Value
Fuel damage	none
RCS activity	
Equilibrium iodine case	0.2 μCi/gm D.E.I-131
Pre-incident iodine spike case	4.0 μCi/gm D.E.I-131
Mass release	1.4E+5 lbm
Steam flashing fraction	0.4
Break isolation time, seconds	5.5
Release mode	Instantaneous as a ground-level release
Plateout, holdup, or dilution credit	none
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Table 10 and 14
CREV system operation	no
CR isolated	no
CR atmospheric dispersion factors	Table 10 and 14

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
for
Fuel Handling Accident

Parameter	Value
Reactor power, MWt,	3016
Radial peaking factor	1.7
Fuel decay period, hours	24
Number of damaged fuel rods	111
Equivalent to damaged fuel assemblies	2.3
Fraction of Gap Activity Released from Damaged Rods	1.0
Fraction of Core Inventory in Gap	
I-131	0.08
Kr-85	0.10
Other halogens and noble gases	0.05
Pool effective decontamination factor	135
Pool water depth	19 feet
Iodine species fraction above pool water	
Elemental	0.57
Organic	0.43
Release duration, hours	
From fuel and pool	Instantaneous
From secondary containment	2
Collection and filtration by SGTS	None
Release point	Reactor building vent stack
Dose conversion factors	FGR11 and FGR12
Atmospheric dispersion factors	Tables 9 and 13
CREVS initiated	no
CR isolated	no
Total assumed CRE intake during CR normal ventilation system operation	
Dresden	6.40+4 cfm
Quad Cities	5.83E+4 cfm
CR normal ventilation system flow rate	2000 cfm (+/-10%)
CR atmospheric dispersion factors	Tables 9 and 13

Table 7

Dresden Units 2 and 3
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for LOCA

Ground level release via MSIV
Elevated release from Containment & ESF via the Station Chimney

Receptor	Time interval	Ground level χ/Q	Elevated χ/Q
EAB	0 - 0.5 hr	---	8.74×10^{-5}
	0.5 - 2 hr	---	6.74×10^{-6}
	0 - 2 hr	2.51×10^{-4}	---
LPZ	0 - 0.5 hr	---	1.55×10^{-5}
	0.5 - 2 hr	---	8.30×10^{-6} Note 1
	0 - 2 hr	2.63×10^{-5}	---
	2 - 8 hr	1.09×10^{-5}	3.57×10^{-6}
	8 - 24 hr	7.02×10^{-6}	2.34×10^{-6}
	24 - 96 hr	2.70×10^{-6}	9.39×10^{-7}
	96 - 720 hr	6.86×10^{-7}	2.53×10^{-7}
Control Room	0 - 2 hr	1.30×10^{-3}	6.42×10^{-6}
	2 - 8 hr	1.06×10^{-3}	2.87×10^{-6}
	8 - 24 hr	4.49×10^{-4}	1.92×10^{-6}
	24 - 96 hr	2.96×10^{-4}	8.03×10^{-7}
	96 - 720 hr	2.44×10^{-4}	2.29×10^{-7}

Note 1: Exelon estimated terrain height inputs for the EAB χ/Q calculations from topographic maps. For the LPZ, Exelon calculated the limiting terrain case by inputting the height of the top of the station chimney as the terrain heights thus reducing the effective release height to ground level. As a result, the 0.5-2-hour LPZ elevated release χ/Q value is higher than 0.5-2-hour elevated release EAB χ/Q value.

Table 8

Dresden Units 2 and 3
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for CRDA

Ground level release via MSIV release pathway

Time interval	EAB χ/Q	LPZ χ/Q	Control room χ/Q
0 – 2 hr	2.51×10^{-4}	2.63×10^{-5}	1.30×10^{-3}
2 - 8 hr	---	1.09×10^{-5}	1.06×10^{-3}
8 – 24 hr	---	7.02×10^{-6}	4.49×10^{-4}

Table 9

Dresden Units 2 and 3
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for FHA

Ground level release via reactor building vent

Receptor	Time interval	χ/Q value
EAB	0 - 2 hr	2.51×10^{-4}
LPZ	0 - 2 hr	2.63×10^{-5}
Control Room	0 - 2 hr	6.44×10^{-4}

Table 10

Dresden Units 2 and 3
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for MSLB

Ground level release

Receptor	Time interval	χ/Q value
EAB	0 - 2 hrs	4.40×10^{-4}
LPZ	0 - 2 hrs	5.50×10^{-5}
Control room	0 - 2 hrs	No χ/Q values were used in dose assessment

Table 11

Quad Cities Units 1 and 2
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for LOCA

Ground level release via MSIV
Elevated release from Containment & ESF via the Station Chimney

Receptor	Time interval	Ground level χ/Q	Elevated χ/Q
EAB	0 - 0.5 hr	---	1.57×10^{-4}
	0.5 - 2 hr	---	6.38×10^{-6}
	0 - 2 hr	1.36×10^{-3}	---
LPZ	0 - 0.5 hr	---	3.01×10^{-5}
	0.5 - 2 hr	---	2.05×10^{-5} Note 2
	0 - 2 hr	1.04×10^{-4}	---
	2 - 8 hr	4.14×10^{-5}	8.76×10^{-6}
	8 - 24 hr	2.62×10^{-5}	5.73×10^{-6}
	24 - 96 hr	9.96×10^{-6}	2.28×10^{-6}
	96 - 720 hr	2.52×10^{-6}	6.07×10^{-7}
Control Room	0 - 2 hr	1.02×10^{-3}	5.84×10^{-6}
	2 - 8 hr	8.23×10^{-4}	2.68×10^{-6}
	8 - 24 hr	3.55×10^{-4}	1.81×10^{-6}
	24 - 96 hr	2.32×10^{-4}	7.77×10^{-7}
	96 - 720 hr	1.38×10^{-4}	2.30×10^{-7}

Note 2: Exelon estimated terrain height inputs for the EAB χ/Q calculations from topographic maps. For the LPZ, Exelon calculated the limiting terrain case by inputting the height of the top of the station chimney as the terrain heights thus reducing the effective release height to ground level. As a result, the 0.5-2 hour LPZ elevated release χ/Q value is higher than 0.5-2 hour elevated release EAB χ/Q value.

Table 12

Quad Cities Units 1 and 2
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for CRDA

Ground level release via MSIV release pathway

Time interval	EAB χ/Q	LPZ χ/Q	Control room χ/Q
0 – 2 hr	1.36×10^{-3}	1.04×10^{-4}	1.02×10^{-3}
2 - 8 hr	---	4.14×10^{-5}	8.23×10^{-4}
8 – 24 hr	---	2.62×10^{-5}	3.55×10^{-4}

Table 13

Quad Cities Units 1 and 2
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for FHA

Ground level release via reactor building vent

Receptor	Time interval	χ/Q value
EAB	0 - 2 hr	1.36×10^{-3}
LPZ	0 - 2 hr	1.04×10^{-4}
Control Room	0 - 2 hr	5.82×10^{-4}

Table 14

Quad Cities Units 1 and 2
Atmospheric Dispersion Factors (χ/Q values in s/m^3) for MSLB

Ground level release via main steam line release pathway

Receptor	Time interval	χ/Q value
EAB	0 - 2 hrs	8.64×10^{-4}
LPZ	0 - 2 hrs	8.69×10^{-5}
Control room	0 - 2 hrs	No χ/Q values were used in dose assessment

(2) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear materials as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

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

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

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

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2957 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fourteenth refueling outage (D3R14).

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state power levels not in excess of 2957 megawatts (thermal), except that the licensee shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in the licensee's telegrams; dated February 26, 1971, have been verified in writing by the Commission.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Restrictions

Operation in the coastdown mode is permitted to 40% power.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

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

D. Equalizer Valve Restriction

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

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined sets of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0," submitted by letter dated October 21, 2004.

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

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

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 229, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

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

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

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