September 8, 2006

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

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ALTERNATIVE SOURCE TERM (TAC NOS. MC6221, MC6222, MC6223, AND MC6224)

Dear Mr. Crane:

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. NPF-37 and Amendment No. 147 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 140 to Facility Operating License No. NPF-72 and Amendment No. 140 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated February 15, 2005, as supplemented by letters dated November 28 and December 9 (two letters), 2005, and January 27, February 13, March 17 and July 14, 2006.

The amendments will fully implement an alternative source term, pursuant to Title 10 of the *Code of Federal Regulations* Section 50.67, and make changes to technical specifications to change plant requirements during movement of fuel that has decayed for a defined time period, increase the maximum containment leakage rate, increase allowable emergency core cooling systems leakage, and change control room ventilation system and filter surveillance acceptance criteria.

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/
Robert F. Kuntz, Project Manager
Licensing Plant Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 147 to NPF-37
2. Amendment No. 147 to NPF-66
3. Amendment No. 140 to NPF-72
4. Amendment No. 140 to NPF-77
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/
Robert F. Kuntz, Project Manager
Licensing Plant Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 147 to NPF-37
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3. Amendment No. 140 to NPF-72
4. Amendment No. 140 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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R. Harvey, NRR	K. Parczewski, NRR	H. Walker, NRR	
M. Hart, NRR	C. Brown, NRR	RidsNrrDraAadb	

Package: ML062340434
Amendment: ML062340420
Tech Spec Pages: ML062550046

*See SE Dated 8/17/06 **See SE dated 2/6/2006 ***TTjader for ****See SE dated 9/6/2005 *****See SE dated 1/18/2006

OFFICE	PM:LPL3-2	LA:LPL3-2	BC:AADB	BC:SCVB	BC:SNPB	BC:ITSB	BC:EICB	BC:CSGB	OGC	BC:LPL3-2
NAME	RKuntz:mw	DClarke	MKotzalas	RDenning*	FAkstulewicz**	TKobetz***	AHowe****	AHiser*****	DRoth	DCollins
DATE	8/25/2006	8/25/2006	8/23/2006	8/17/2006	2/6/2006	8/24/2006	9/6/2005	1/18/2006	9/6/2006	9/8/2006

OFFICIAL RECORD COPY

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EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. NPF-37

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 15, 2005, as supplemented by letters dated November 28 and December 9, 2005 (two letters), and January 27, February 13, March 17 and July 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.147 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 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 Facility Operating License

Date of Issuance: September 8, 2006

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. NPF-66

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 15, 2005, as supplemented by letters dated November 28 and December 9, 2005 (two letters), and January 27, February 13, March 17 and July 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No.147 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 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 Facility Operating License

Date of Issuance: September 8, 2006

ATTACHMENT TO LICENSE AMENDMENT NOS. 147 AND 147

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

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 1 License Page 3	Unit 1 License Page 3
Unit 2 License Page 3	Unit 2 License Page 3
1.1-3	1.1-3
1.1-6	1.1-6
3.3.7-2	3.3.7-2
3.3.7-3	3.3.7-3
3.3.7-4	--------
3.3.8-2	3.3.8-2
3.3.8-4	3.3.8-4
3.7.10-2	3.7.10-2
3.7.10-3	3.7.10-3
3.7.11-2	3.7.11-2
3.7.11-3	3.7.11-3
3.7.13-1	3.7.13-1
3.7.13-2	3.7.13-2
3.7.13-3	3.7.13-3
3.7.13-4	3.7.13-4
3.9.4-1	3.9.4-1
3.9.4-2	3.9.4-2
3.9.7-1	3.9.7-1
5.5-17	5.5-17
5.5-18	5.5-18
5.5-24	5.5-24

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. NPF-72

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 15, 2005, as supplemented by letters dated November 28 and December 9, 2005 (two letters), and January 27, February 13, March 17 and July 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 140 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 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 Facility Operating License

Date of Issuance: September 8, 2006

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. NPF-77

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated February 15, 2005, as supplemented by letters dated November 28 and December 9, 2005 (two letters), and January 27, February 13, March 17 and July 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 140 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance and shall be implemented within 120 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 Facility Operating License

Date of Issuance: September 8, 2006

ATTACHMENT TO LICENSE AMENDMENT NOS. 140 AND 140

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

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 1 License Page 3	Unit 1 License Page 3
Unit 2 License Page 3	Unit 2 License Page 3
1.1-3	1.1-3
1.1-6	1.1-6
3.3.7-2	3.3.7-2
3.3.7-3	3.3.7-3
3.3.7-4	--------
3.3.8-2	3.3.8-2
3.3.8-4	3.3.8-4
3.7.10-2	3.7.10-2
3.7.10-3	3.7.10-3
3.7.11-2	3.7.11-2
3.7.11-3	3.7.11-3
3.7.13-1	3.7.13-1
3.7.13-2	3.7.13-2
3.7.13-3	3.7.13-3
3.7.13-4	3.7.13-4
3.9.4-1	3.9.4-1
3.9.4-2	3.9.4-2
3.9.7-1	3.9.7-1
5.5-17	5.5-17
5.5-18	5.5-18
5.5-24	5.5-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated February 15, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML050560102), as supplemented by letters dated November 28 (ADAMS Accession Number ML053330025), and December 9, 2005 (ADAMS Accession Numbers ML060040046 and ML060040081), and January 27 (ADAMS Accession Number ML0060300381), February 13 (ADAMS Accession Number ML060450176), March 17 (ADAMS Accession Number ML060790055) and July 14, 2006 (ADAMS Accession Number ML061950332), Exelon Generation Company, LLC (Exelon, the licensee) requested changes to the technical specifications (TSs) and surveillance requirements (SRs) for the Byron Station, Unit Nos. 1 and 2 (Byron), and Braidwood Station, Unit Nos. 1 and 2 (Braidwood). The November 28 and December 9, 2005 (two letters), and January 27, February 13, March 17 and July 14, 2006, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards considerations.

The proposed changes would fully implement an alternative source term (AST), pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 "Accident source term," and make changes to TSs to change plant requirements during movement of fuel that has decayed for a defined time period, increase the maximum containment leakage rate, increase allowable emergency core cooling systems (ECCS) leakage, and change control room (CR) ventilation system and filter surveillance acceptance criteria.

To support a full-scope implementation of the AST methodology, radiological consequence analyses were performed for the following six bounding design-basis accidents (DBAs) that result in CR and offsite exposure.

- loss-of-coolant accident (LOCA)

- fuel-handling accidents (FHA) in the fuel handling building (FHB) and in containment

- control rod ejection accident (CREA)

- locked rotor accident (LRA)

- main steam line break (MSLB)

- steam generator tube rupture (SGTR)

Proposed changes to the current licensing basis for Byron and Braidwood by AST analyses include the following:

- a change in the definition for DOSE EQUIVALENT I-131 to reflect the use of exposure-to-dose conversion factors for inhalation from Federal Guidance Report 11 (FGR-11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988;

- addition of a new definition for RECENTLY IRRADIATED FUEL, i.e., fuel that has occupied part of a critical reactor core within the previous 48 hours;

- the Containment Leakage Rate Testing Program maximum allowable containment leakage rate, (i.e., L_a) at design accident pressure (i.e., P_a) is increased from 0.1 percent to 0.2 percent of containment air weight per day;

- ECCS allowable leakage is assumed to be 276,000 cubic centimeters per hour (cc/hr) for the Byron and Braidwood units, rather than 15,249 cc/hr (approximately 4.0 gallons per hour (gal/hr)) for Braidwood and 13,294 cc/hr (approximately 3.5 gal/hr) for Byron; and

- the CR ventilation filtration system makeup filter charcoal adsorber bypass acceptance criterion is increased from 0.05 percent to 1 percent and the associated charcoal adsorber penetration acceptance criterion is increased from 0.5 percent to 2.0 percent. These changes reflect a reduction in the assumed filtration efficiency from 99 percent to 95 percent. This penetration acceptance criteria maintains a safety factor of two consistent with the guidance in NRC Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," June 3, 1999.

TS and associated Bases revisions were proposed to change the applicability requirements for the below systems to be applicable only during movement of RECENTLY IRRADIATED FUEL assemblies and to relax the operability requirements of these systems during core alterations, unless required for other reasons.

- containment
- FHB ventilation system

The licensee proposed the following TS changes:

TS Section 1.1, "Definitions," DOSE EQUIVALENT I-131

The proposed change revises the definition of DOSE EQUIVALENT I-131 to remove the word "thyroid" and add a reference to FGR-11, Table 2.1, "Exposure-to-Dose Conversion Factors for Inhalation."

TS Section 1.1, "Definitions" RECENTLY IRRADIATED FUEL

The proposed change adds the definition of RECENTLY IRRADIATED FUEL. RECENTLY IRRADIATED FUEL is defined as: "Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a critical reactor core is referred to as irradiated fuel."

TS 3.3.7, "VC Filtration System Actuation Instrumentation"

Required Action E.1, "Suspend CORE ALTERATIONS" and its associated completion time are deleted consistent with the guidance in Technical Specification Task Force Traveler (TSTF)-51.

TS 3.3.8, "FHB Ventilation System Actuation Instrumentation"

The words "irradiated fuel" in Required Actions B.2.1 and B.2.2 are replaced with the defined term RECENTLY IRRADIATED FUEL. Required Action B.2.3 is deleted consistent with the guidance in TSTF-51.

TS Table 3.3.8-1, "FHB Ventilation System Actuation Instrumentation"

In Notes (a) and (b), the term "irradiated fuel" is replaced with the defined term RECENTLY IRRADIATED FUEL. Note c), applicable to Function 1, "Fuel Handling Building Radiation" is deleted, as well as the note itself, consistent with the guidance in TSTF-51 and supported by the AST analysis. Deletion of the note eliminates the requirement to have the FHB radiation monitors operable during CORE ALTERATIONS with the equipment hatch not intact.

TS 3.7.10, "VC Filtration System"

Required Action C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions C.2.2 and C.2.3 are renumbered to support deletion of Required Action C.2.1. This is done in accordance with the guidance in TSTF-51 and is supported by the AST analysis.

Required Action D.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions D.2 and D.3 are renumbered to support deletion of Required Action D.1.

TS 3.7.10, "VC Filtration System," SR 3.7.10.4

NOTE: The proposed change to TS SR 3.7.10.4 was withdrawn from consideration by Exelon in its letter dated March 17, 2006.

The words "the upper cable spreading room at positive pressure of 20.02 inches water gauge and" are removed. The word "area" (i.e., in the statement, ". . . adjacent to the control room *area* during . . . ") is replaced with the word "envelope." This change is consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," and is further justified by the requirements of the proposed Byron and Braidwood Control Room Envelope Integrity Program, which verifies that the control room envelope is maintained at a positive pressure.

TS 3.7.11, "VC Temperature Control System"

Required Action C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions C.2.2 and C.2.3 are renumbered to support deletion of Required Action C.2.1.

Required Action D.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions D.2 and D.3 are renumbered to support deletion of Required Action D.1.

TS 3.7.13, "FHB Ventilation System"

The term "irradiated fuel" in the applicability statement is replaced with the words "RECENTLY IRRADIATED FUEL," in accordance with the guidance in TSTF-51. The reference to suspension of CORE ALTERATIONS is deleted.

The term "irradiated fuel" in Required Actions B.2.1 and B.2.2 is replaced with the term "RECENTLY IRRADIATED FUEL." Required Action B.2.3 regarding the suspension of CORE ALTERATIONS is deleted.

The term "irradiated fuel" in Required Actions C.1 and C.2 is replaced with the words "RECENTLY IRRADIATED FUEL." Required Action C.3 regarding the suspension of CORE ALTERATIONS is deleted.

In the NOTE to SR 3.7.13.3, the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL, and the words "or CORE ALTERATIONS," are deleted in accordance with the guidance in TSTF-51.

In the NOTE to SR 3.7.13.5, the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL in accordance with the guidance in TSTF-51.

TS 3.9.4, "Containment Penetrations"

The reference to "LCO [Limiting Condition for Operation]" in the NOTE is changed to "TS." This is an editorial change for clarity.

NOTE: The proposed revision to the NOTE in TS 3.9.4 was changed by the licensee's letter dated July 14, 2006 (see section 3.9.9 of this SE.)

The term "irradiated fuel" in the applicability statement is replaced with the words "RECENTLY IRRADIATED FUEL." The reference to the suspension of CORE ALTERATIONS is deleted. In Required Action A.1, the reference to the suspension of CORE ALTERATIONS and its associated completion time are deleted. Required Action A.2 is renumbered and the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL in accordance with the guidance in TSTF-51.

TS 3.9.7, "Refueling Cavity Water Level"

The reference in the applicability statement to the suspension of CORE ALTERATIONS is deleted; however, the reference to "irradiated fuel assemblies" is retained in this location in accordance with TSTF-51.

In Required Action A.1, the reference to the suspension of CORE ALTERATIONS and its associated completion time are deleted in accordance with TSTF-51 supported by the AST analysis. Required Action A.2 is renumbered to A.1 to support the deletion of Required Action A.1.

TS 5.5.11, "Ventilation Filter Testing Program (VFTP)"

In Item b, the charcoal adsorber bypass acceptance criterion is changed from 0.05 percent to 1 percent. In Item c, the charcoal adsorber penetration acceptance criterion is changed from 0.5 percent to 2 percent. These changes reflect a reduction in assumed filtration efficiency from 99 percent to 95 percent, used in the AST analysis, and maintains a safety factor of 2 for the penetration acceptance criterion.

TS 5.5.16, "Containment Leakage Rate Testing Program"

The maximum allowable containment leakage rate leakage limit (L_a) is changed from 0.1 percent to 0.2 percent of containment air weight per day. This change is supported by the AST dose consequence analyses.

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee, in Section 5.2 of its February 15, 2005, submittal, identified the applicable regulatory requirements. The NRC's traditional methods for calculating the radiological consequences of DBAs (i.e., prior to adopting the AST methodology) are described in a series of Regulatory Guides (RGs) and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the Technical Information Document TID-14844 source term and the whole-body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance." Many of those analysis assumptions and methods are inconsistent with the AST methodology and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67.

RG 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" provides assumptions and methods that are acceptable to the NRC staff for performing design-basis radiological analyses using an AST approach. This guidance supersedes corresponding radiological analysis assumptions provided in the previous RGs and SRP chapters when used in conjunction with an approved AST methodology and the TEDE criteria provided in 10 CFR 50.67.

Due to the comprehensive nature of RG 1.183, Attachment 7 to the licensee's February 15, 2005 submittal, "Regulatory Guide Conformance Tables," (i.e., Tables A-G) was developed to show how each section of the RG 1.183 guidance is being addressed. In addition, Table H of Attachment 7 shows conformance with the diffuse area source guidance in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," December 2001.

The NRC also published a new SRP section to address AST, SRP Section 15.0.1, Revision 0, "Radiological Consequence Analyses Using Altemative Source Terms," which is consistent with the guidance found in RG 1.183.

The regulatory requirements for CR dose are found in 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control room." As guidance, the NRC staff consulted Sections 6.4, "Control Room Habitability System," and 9.4.1, "Control Room Area Ventilation System," of NUREG-0800, "Standard Review Plan" (SRP), GL 99-02 and 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." The NRC staff also considered the relevant information in Byron and Braidwood Updated Final Safety Analysis Report (UFSAR).

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment which are described in Sections 4 and 5 of the licensee's February 15, 2005, submittal. The detailed evaluation below will support the conclusion 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.1 Atmospheric Dispersion Estimates

The licensee calculated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of the DBAs on the CR and offsite exposures. The licensee used the ARCON96 and PAVAN atmospheric dispersion computer models to calculate χ/Q values for the LOCA, FHA in the FHB and in containment, CREA, LRA, MSLB accident, and SGTR accident. The resulting CR and offsite (exclusion area boundary (EAB) and low population zone (LPZ)) χ/Q values represent a change from those currently presented in the Byron and Braidwood UFSARs.

3.1.1 Meteorological Data

The licensee generated the new CR and offsite χ/Q values for this amendment application using site meteorological data collected at both Byron and Braidwood during the period of 1994 through 1998. The licensee provided these data in the form of hourly data files (for input into the ARCON96 computer code) and joint wind speed, wind direction, and atmospheric stability frequency distributions (for input to the PAVAN computer code) in Attachment 8 to its February 15, 2005, application.

Wind measurements at Braidwood were taken at 34 feet (10.4 meters) and 203 feet (61.9 meters) above ground level (AGL) and the vertical temperature difference was measured between the 199-foot (60.7-meter) and 30-foot (9.1-meter) levels. Byron wind measurements were taken at 30 feet (9.1 meters) and 250 feet (76.2 meters) AGL and the vertical temperature difference measured between 250 feet (76.2 meters) and 30 feet (9.1 meters). Vertical temperature difference (delta-T) was the method used to determine atmospheric stability.(1)

The licensee stated that the Braidwood and Byron meteorological towers are equipped with instrumentation that conforms with the system accuracy recommendations of RG 1.23, "Onsite Meteorological Programs." The licensee also stated that instrument calibrations and data consistency evaluations were performed routinely to ensure data integrity. The combined data recovery of wind speed, wind direction, and stability data exceeded the RG 1.23 goal of 90 percent for each of the 5 years (1994 through 1998) for both the Byron and Braidwood sites.

The NRC staff performed a quality review of the 1994 through 1998 hourly meteorological database provided by the licensee using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. Wind speed, wind direction, and stability class frequency distributions were reasonably similar from year to year and the 1994 through 1998 frequency distributions were reasonably consistent with the 1974 through 1976 data presented in Section 2.3 of the Byron and Braidwood UFSARs, with the exception of Braidwood's lower level wind direction data. The prevailing Braidwood lower level 1974 through 1976 wind direction sectors were S and SSW whereas W and WNW were the prevailing wind directions during 1994 through 1998(2). With respect to atmospheric stability, the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions

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(1)The licencee stated in Attachment 1 to its February 15, 2005, submittal that delta-T was the principal method for determining atmospheric stability and standard deviation of the horizontal wind direction (sigma-theta) was a backup method used when delta-T data were not available. In its letter dated February 13, 2006, the licensee clarified that delta-T was the only method used to determine atmospheric stability for the data files used in this amendment application. Substitution of delta-T data with sigma-theta data was not required because of acceptable data recovery rates.

(2)The licensee addressed the shift in the Braidwood lower level wind data in its February 13, 2006 letter. According to the licensee, there were no changes in the meteorological tower's structure or instrument boom orientation between the 1974 through 1976 and 1994 through 1998 data sets. There were also no obvious instrumentation issues that could account for the wind direction shift and there is no significant vegetation in the immediate vicinity of the meteorological monitoring tower. The licensee concluded that natural climatological variability cannot be excluded as a significant contributor. The licensee considered the 1994 through 1998 wind direction sector frequencies to be more consistent with what would be anticipated based upon the geographic location and topography and concluded that the Braidwood 1994 through 1998 lower level wind direction data set is more representative of current conditions than the 1974 through 1976 data set.

occurring during the day). A comparison of joint frequency distributions (JFDs) derived by the NRC staff from the ARCON96 hourly data with the JFDs developed by the licensee for input into the PAVAN atmospheric dispersion model showed reasonable agreement.

For the reasons cited above, the NRC staff concludes that the 1994-1998 Byron and Braidwood site meteorological databases provide an acceptable basis for making atmospheric dispersion estimates for use in the DBA dose assessments performed in support of this application for amendment.

3.1.2 Control Room Atmospheric Dispersion Factors

The Byron and Braidwood CRs are common facilities which serve both Units 1 and 2 at their respective sites. Each CR facility is served by two completely redundant heating ventilation and air-conditioning equipment trains and each equipment train has two air intakes: a normal air intake and a turbine building emergency air intake. During the CR normal mode of operation, outside air is introduced into the CR through the normal air intake to maintain positive pressure with respect to surrounding areas and to makeup for air that is exhausted. Upon detection of high radiation in the normal air intake or upon a safety injection signal, the normal air intake dampers are automatically closed and outside air is introduced into the CR through the normally closed dampers of the emergency air intake.

The licensee calculated CR χ/Q values for four release locations at each unit at Byron and Braidwood. The four release locations were (1) containment wall (i.e., containment leakage), (2) plant vent, (3) steam generator power operated relief valves (SG PORVs) and safety valves, and (4) MSLB. For each potential release location, the licensee used the worst case combination of χ/Q values from all four units at Byron and Braidwood to determine χ/Q values for both the normal air intakes and the emergency air intakes. Thus, for each time interval, the worst case χ/Q value was applied to both sites, irrespective of the site for which it was actually calculated. The licensee assumed the activity released to the environment during the initial 30 minutes of each DBA is introduced to the CR via the normal air intake with the highest χ/Q values. After this period, when the normal air intake is assumed to be closed, the licensee assumed filtered air makeup is introduced to the CR via the emergency air intake with the highest χ/Q values. Unfiltered inleakage was also assumed to occur throughout the duration of each event and was modeled using the normal air intake χ/Q values during the initial 30 minutes and the emergency air intake χ/Q values thereafter.[3]

The licensee used guidance provided in RG 1.194 to generate the new CR atmospheric dispersion factors. The licensee calculated these new control room χ/Q values 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.

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[3]Regarding its unfiltered inleakage assumptions, the licensee stated in its February 13, 2006, letter that site walkdowns, drawing reviews, and CR tracer gas test results have confirmed that there are no potential unfiltered inleakage pathways during either normal or emergency operating modes that could result in χ/Q values higher than the CR air intake χ/Q values used, with the exception of isolation dampers associated with the normal air intake. In the emergency operating mode, inleakage through these isolation dampers would be filtered by the recirculation filters. As a result, treating this inleakage as unfiltered with the emergency intake χ/Q values is more conservative than as filtered with the normal air χ/Q values.

The licensee executed ARCON96 using the 1994-1998 hourly data from the Byron and Braidwood site meteorological towers. Wind speed and direction from both tower levels were provided as input. Because the heights of all four release locations are less than 2½ times the height of adjacent buildings, all four release locations were modeled using the ARCON96 ground level release option in accordance with RG 1.194.

The resulting CR χ/Q values used in the dose analyses are presented in Table 1. Details on the licensee's assessments of CR post-accident atmospheric dispersion conditions for each release location are provided below:

a. Containment Leakage Releases: The containment leakage pathway was used by the licensee to model releases associated with (1) LOCA and (2) CREA where the ejected control rod is assumed to breach the reactor pressure vessel (RPV), effectively causing the equivalent of a small break LOCA. The licensee modeled the Byron and Braidwood containment leakage releases as area (diffuse) sources. The area source dimensions were the maximum vertical and horizontal dimensions of the above-grade containment building cross-sectional area perpendicular to the line of sight from the containment building center to the CR air intakes. The initial diffusion coefficients (i.e., initial σ_y and σ_z values) were determined by dividing the area source dimensions by a factor of six in accordance with RG 1.194. The release heights were set equal to the mid-height of containment buildings above plant grade (APG) (i.e., 29.7 meters APG) as compared to the normal air intake height of 21.2 meters APG and the emergency air intake height of 20.4 meters APG. The resulting ARCON96 analyses showed that the Braidwood, Unit 1 containment leakage release to both the Braidwood, Unit 1 normal air intake and the Braidwood, Unit 1 emergency air intake were the bounding source-receptor combinations that resulted in the highest (i.e., most conservative) χ/Q values.

b. Plant Vent Releases: Each of the Byron and Braidwood units has a main plant vent. The plant vent pathway was used by the licensee to model releases associated with (1) LOCA ECCS leakage carrying reactor coolant outside containment into the auxiliary building and (2) the FHA for accidents occurring both in containment[(4)] and in the FHB.[(5)] The licensee modeled the Byron and Braidwood plant vent releases as point sources. The plant vent release heights are 61 meters APG as compared to the normal air intake height of 21.2 meters APG and the emergency air intake height of 20.4 meters APG. The resulting ARCON96 analyses showed that the bounding plant vent release to normal air intake source-receptor combination was the Braidwood, Unit 2 plant vent release to the Braidwood, Unit 2 normal air intake. The resulting bounding plant vent release to emergency air intake source-receptor combination varied as a function of averaging time period. The licensee used a conservative composite of these source-receptor combinations to derive the plant vent release χ/Q values.

[(4)]The licensee evaluated alternative containment release points through major openings such as the personnel/equipment hatch to the outside and concluded that the plant vent χ/Q values were bounding.

[(5)]The licensee evaluated alternative FHB release points through major openings such as a inner or outer trackway roll-up door and concluded that the plant vent χ/Q values were bounding.

c. SG PORV and Safety Valve Releases: Byron and Braidwood each have four main steam lines for each unit. Each steam line has a set of five safety valves and one SG PORV. The SG PORV valves operate at a lower pressure point than the safety valves to reduce unnecessary operation of the safety valves. The SG PORV/safety valve pathway was used by the licensee to model releases associated with (1) the CREA where the reactor coolant system (RCS) integrity is maintained and all activity within the RCS is assumed to leak through the SG tubes, (2) LRA, (3) intact SG loops for the MSLB accident, and (4) SGTR accident.

The licensee modeled the SG PORV/safety valve releases as point sources. The SG PORV/safety valve release heights are 9.8 meters APG as compared to the normal air intake height of 21.2 meters APG and the emergency air intake height of 20.4 meters APG. The resulting ARCON96 analysis showed that the bounding source-receptor combination varied as a function of averaging time period. The licensee used a conservative composite of source-receptor combinations to derive the SG PORV/safety valve χ/Q values.

RG 1.194 allows the ground level χ/Q values calculated with ARCON96 (based on the physical height of the release point) to be reduced by a factor of 5 if (1) the release point is uncapped and vertically oriented and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed by a factor of 5. SG PORV/safety valve releases satisfy the first criterion in that the release points are uncapped and vertically oriented. The licensee stated that the second criterion is met based on their engineering judgment that the SG PORV/safety valve releases exceed the 95th percentile wind speed due to choked sonic flow conditions[(6)]. Consequently, the licensee reduced the resulting ARCON96 SG PORV/safety valve χ/Q values by a factor of 5.

d. MSLB: Byron and Braidwood each have four main steam lines for each unit. The MSLB pathway was used by the licensee to model the broken steam line releases associated with the MSLB accident. The licensee modeled the Byron and Braidwood MSLB releases as point sources. For the normal air intake dispersion analyses, the licensee used a “taut string length” around the intervening auxiliary building structure to determine the horizontal distance between the release point and intake and set the intake height equal to the release height of 7.9 meters APG. The resulting ARCON96 analysis showed that the bounding source-receptor combinations varied as a function of averaging time period. The licensee used a conservative composite of source-receptor combinations to derive the MSLB χ/Q values.

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of the ARCON96 model for the Byron and Braidwood sites. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and site practice. The NRC staff made an independent evaluation of the resulting bounding

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[(6)]Both the LRA and SGTR analyses assume a 20-minute SG PORV failure in the open position at the onset of the accident releases. In its February 13, 2006, letter, the licensee stated that transient thermal-hydraulic analyses confirmed choked flow conditions exist throughout the pressure ranges associated with the SGTR and LRA. The licensee also stated that, consistent with its current licensing basis, a failed SG PORV is not assumed for the MSLB and CREA analyses.

atmospheric dispersion estimates by running the ARCON96 computer code and obtaining similar results. On the basis of this review, the NRC staff concludes that the χ/Q values for DBA releases to the Byron and Braidwood CRs as presented in Table 1 are acceptable for use in the DBA CR dose assessments performed in support of this amendment application.

3.1.3 Offsite Atmospheric Dispersion Factors

The licensee used the NRC-sponsored computer code PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") to generate new χ/Q values for the EAB and LPZ. The PAVAN model implements the methodology outlined in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants."

The licensee executed PAVAN twice: once based on Braidwood onsite meteorological data and distances to the EAB and LPZ, and a second time based on Byron onsite meteorological data and EAB and LPZ distances. The licensee's meteorological input to PAVAN consisted of a JFD of wind speed, wind direction, and atmospheric stability data for the period 1994-1998. Wind data from the lower tower levels were used. The licensee's JFD consisted of seven wind speed categories in accordance with RG 1.23. The licensee's JFD showed that approximately 12 percent of the Braidwood wind speed data and 16 percent of the Byron wind speed data fell within the first wind speed class which ranged from 0.8 miles per hour (0.36 meters per second) to 3.5 miles per hour (1.56 meters per second). The PAVAN atmospheric dispersion computer code produces the best results for 5 percentile χ/Q values if the wind speed data are classified into a large number of categories at the lower wind speeds, thus preventing the data from being clustered into a few categories. Consequently, the NRC staff reran PAVAN using a finer wind speed category breakdown for the lower wind speeds and calculated more conservative χ/Q values[(7)]. In response, the licensee committed in its letter dated February 13, 2006, to update its calculation of χ/Q values using finer wind speed categories provided in the latest appropriate regulatory guidance the next time the dose consequence calculations associated with the LOCA, MSLB, CREA, LRA, SGTR, and FHA events are revised[(8)]. Nonetheless, there is

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[(7)]The NRC staff reran PAVAN using a larger number of low wind speed categories (i.e., 0.5, 0.75, 1.0, 1.5, 2.0, 3.0, 5.0, and 10 meters per second) based on the wind speed categories discussed in Section 4.6 of NUREG/CR-2858.

[(8)]Subsequent to the licensee's February 13, 2006, the NRC staff issued Regulatory Issue Summary (RIS) 2006-04 on March 7, 2006, entitled, "Experience with Implementation of Alternative Source Terms." RIS 2006-04 discusses, among other items, the use of a large number of wind speed categories at the lower wind speeds for the JFDs used as input to the PAVAN computer code.

significant margin between the licensee's calculated EAB/LPZ doses and the corresponding dose limits to utilize the licensee's calculated EAB/LPZ χ/Q values for this license amendment[9].

All EAB releases at each site were assumed to occur through the outer containment wall, resulting in EAB distances of 485 meters for the Braidwood site and 445 meters for the Byron site. All LPZ releases at each site were assumed to occur at the midpoint between the two reactors, resulting in LPZ distances of 1,810 meters for the Braidwood site and 4,828 meters for the Byron site. Because the heights of all release locations are less than 2½ times the height of adjacent buildings, all releases were assumed to be ground level (e.g., release heights were set to 10 meters APG). Building wake effects were modeled based on a containment building vertical cross-sectional area of 2,917 square meters.

The PAVAN analyses showed that the Byron site had the highest (i.e., most conservative) EAB χ/Q values and the Braidwood site had the highest LPZ χ/Q values. The resulting bounding EAB and LPZ χ/Q values from each site were used at both sites and are presented in Table 2.

The NRC staff evaluated the applicability of the PAVAN model and concluded that there are no unusual release characteristics, plant configuration, or site topography that preclude the use of the PAVAN model for the Byron and Braidwood sites. The NRC staff qualitatively reviewed the inputs to the PAVAN calculations and, except as noted above, found them generally consistent with site configuration drawings and site practice. On the basis of this review, the NRC staff concludes that the EAB and LPZ χ/Q values presented in Table 2 are acceptable for use in the DBA offsite dose assessments performed in support of this amendment application. Note that the licensee has committed to update its calculation of χ/Q values using finer wind speed categories provided in the latest appropriate regulatory guidance the next time it revises the dose calculations associated with this license amendment (i.e., the dose calculations associated with the LOCA, MSLB, CREA, LRA, SGTR, and FHA events). The NRC staff has subsequently issued RIS 2006-04 which, among other items, discusses the use of a large number of wind speed categories at the lower wind speed for JFDs used as input to the PAVAN atmospheric dispersion model.

3.2 LOCA Radiological Consequences Analysis

Exelon performed an analysis of the radiological consequences of the design-basis LOCA using an AST. To show compliance with 10 CFR 50.67, the licensee calculated the maximum TEDE at the EAB for any 2-hour period, the TEDE at the boundary of the LPZ over the duration of the accident, and the TEDE in the CR over the duration of the accident. The NRC staff reviewed the description of the analysis as submitted and finds that the licensee followed the guidance in RG 1.183 regarding AST calculation methodologies. A discussion of the NRC staff's review of the licensee's analysis, with emphasis on exceptions to the guidance, follows.

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[9]The licensee presented a table on page 8 of Attachment 1 to its letter dated February 13, 2006, comparing the EAB/LPZ calculated doses to the corresponding dose limit for each accident. This table shows that the accident with the smallest margin (percentage-wise) is the CREA at the EAB (e.g., a 36 percent increase in CREA doses at the EAB would reach the EAB dose limit). The NRC staff's calculated EAB χ/Q value is 30 percent higher than the χ/Q value used by the licensee to calculate CREA doses at the EAB. Therefore, CREA EAB doses re-calculated using the NRC staff's χ/Q value would increase by 30 percent and still be within regulatory limits.

3.2.1 LOCA Source Term

The radiological consequences analysis of the design basis LOCA assumes full core melting, with release of the radioactive material to the RCS and then to the containment over a total period of 1.8 hours. Release of radioactive material to the environment is assumed to occur through leakage from the containment and leakage of containment sump water from ECCS components outside containment after recirculation begins.

The licensee's analysis source term was calculated for a core power level of 3,658.3 megawatts thermal (MWt), which includes a calorimetric measurement uncertainty of 2 percent over the rated thermal power of 3,586.6 MWt. The licensee used ORIGIN 2.1 methodology to determine the core inventory which the NRC finds acceptable in accordance with Section 3.1 of RG 1.183. The licensee's assumptions, with regard to the source-term release fractions, release timing, and radionuclide composition, follow the guidance in Regulatory Position 3 of RG 1.183. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the iodine entering the containment from the damaged core during an accident is composed of at least 95 percent cesium iodide (CsI). Upon its dissolution in sump water the iodine will be predominantly in an iodide form (I^-). However, in the radiation field existing in the containment, some of this iodine will be converted into the molecular form (I_2). This conversion is strongly dependent on pH and will increase as the value of pH decreases. Since molecular iodine is scarcely soluble in water, some of it will be released to the containment atmosphere and leak to the outside, contributing to possible radiation dose. However, if the pH of the sump water is maintained at 7 or above, formation of molecular iodine will be impeded and its release to the outside considerably reduced. In the Byron and Braidwood plants this is achieved by adding to the sump water a sodium hydroxide (NaOH) solution. When NaOH is combined with boric acid, a buffered solution is formed. If sufficient NaOH is added, its buffering action will maintain sump water pH above 7, even with the addition of strong acids. Two strong acids are generated in a post-LOCA environment: hydrochloric acid, generated by radiolytic decomposition of Hypalon insulation, and nitric acid, produced by irradiation of water and air in the containment.

After a LOCA, borated water is accumulated in the containment sump from the following sources:

- Refueling Water Storage Tank (RWST): 457,904 gallons at 2500 parts per million Boron (ppmB)
- RCS: 620,800 pounds mass at 2300 ppmB
- Four Accumulators: 28,868 gallons (total) at 2400 ppmB

In order to maintain pH basic, the licensee calculated that 2,500 gallons of 30 weight percent NaOH solution must be added to the sump water. With this amount of NaOH solution, the licensee calculated value of pH immediately after NaOH addition is 8.0. However, in the post-LOCA environment, hydrochloric and nitric acids are continuously formed in the containment and the pH will decrease. The licensee did not perform an independent calculation of this pH change, but used the values calculated in a similar plant (Seabrook Unit 1). The licensee's comparison of the parameters affecting generation of strong acids in the post-LOCA environment of the Byron and Braidwood units and in similar plants has shown very close similarity. This justified the use of the sister plant data by the licensee. The justification was also supported by a very small effect produced by strong acids on pH. With the amount of

strong acids generated in the containment in 30 days after a LOCA, the decrease of sump water pH was less than 0.1 pH unit. The lowest value of sump water pH determined by the licensee was, therefore, 7.9. At this value of pH, according to the NUREG/CR-5950, “Iodine Evolution and pH Control,” methodology, the fraction of radioactive iodine converted from ionic (I^-) to molecular (I_2) form was negligible.

The NRC staff performed an independent evaluation of the sump water pH using different methodology from that of the licensee. The value of pH immediately after addition of the NaOH solution was 8.3 and the pH change due to addition of hydrochloric and nitric acids was less than 0.1 pH unit. The lowest value of pH was 8.2. It was higher than the value determined by the licensee confirming the conservatism existing in the licensee analysis.

After the end of post-LOCA injection phase, the ECCS and containment spray systems are switched to the recirculation phase. This requires that water from the containment sump replace the RWST as the ECCS supply which is accomplished by realigning several system valves. In some cases, such a change may result in opening a leakage path between the containment sump and the RWST through packing glands, pump shaft seals, and flanged connections. The leaking containment water will contain radioactive iodine, but its pH will be buffered to above 7 pH by NaOH. However, when it mixes with the water remaining in the RWST which has boron concentration between 2300 and 2500 ppmB, its pH will drop down to a value significantly lower than 7. This will produce extensive conversion of iodine from an ionic to a molecular form and will result in iodine release. Since the RWST is located outside of the containment, the released iodine will contribute directly to the plant radiation dose.

The licensee provided a detailed description of the Byron and Braidwood EOP 1/2B(w)EP ES-1.3, that will ensure that after the ECCS realignment from the RWST to the containment sump, there will be enough separation provided between the containment sump and the RWST to prevent sump water from leaking into the RWST. The EOP dictates the transfer of the ECCS from the RWST to the containment sump following a specific sequence of valve arrangements that will isolate the ECCS from the RWST. The potential leakage path back to the RWST will be isolated by at least by two valves in series, typically a check valve in combination with a motor operated valve. Only in one case, during the realignment of the residual heat removal (RHR) pump suctions from the RWST to the containment sump, will the back leakage to the RWST be prevented by a single check valve. This situation will last until the RHR pump suction isolation valve in series with the check valve is closed. During this short time required to close the isolation valve, the chances of check valve failure and ingress of containment sump water to RWST is unlikely. Therefore, there is no need to address the issue of pH in the RWST.

The NRC staff reviewed the licensee’s EOP including the sequence of valve operation during realignment of the ECCS to the containment sump. Based on the results of this review, the NRC staff finds that using this EOP the licensee will prevent containment water leakage into the RWST that would potentially result in radioactive iodine release from the RWST to the environment.

3.2.2 LOCA Containment Release

Exelon assumed that the source term is released into the containment in accordance with RG 1.183, Table 2 release timing. The values for containment sprayed and unsprayed volumes and air exchange rate between the two volumes are unchanged from the current licensing and design basis. The containment volume is 2.85 x 10^6 cubic feet with 82.5 percent of the containment volume sprayed. Initial activity distribution is 82.5 percent in the sprayed region, and 17.5 percent in the unsprayed region. Transfer between these two volumes is assumed to be limited to that provided by the containment fan coolers, which are also known as deck fans. Even without the fan coolers, the licensee considers that there would be significant mixing induced by the containment sprays and by the combination of steaming and heat transfer. Since the minimum number of fan coolers assured operable will be 2 out of 4, and the flow rate is 65,000 cubic feet per minute (cfm) per deck fan, the licensee assumed a total of 130,000 cfm flow from the unsprayed region to the sprayed region, and vice versa.

Exelon's analysis took credit for removal of particulate iodine in the containment through natural processes. The Powers Natural Deposition algorithm, based on NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," was used in Exelon's analysis using the RADTRAD computer code (NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation"). No natural deposition of elemental or organic iodine was assumed. The lower bound, or 10th percentile, level of deposition values was used. These assumptions are in accordance with the guidance in RG 1.183 and are acceptable to the NRC staff.

Credit was taken in the containment for removal of fission products other than organic iodine and noble gases by the containment spray system. The methodologies used by Exelon to calculate the time constants for elemental iodine removal and particulate removal by sprays follows the guidance in RG 1.183 and more specific guidance in SRP 6.5.2, Revision 2, "Containment Spray as a Fission Product Cleanup System." The Byron and Braidwood current licensing basis includes spray removal using models from SRP 6.5.2, Revision 2. Credit was taken for elemental iodine removal until a decontamination factor (DF) of 200 was reached. The licensee also assumed that the particulate spray removal was reduced to 10 percent of the calculated value after a DF of 50 was reached. The major differences between the previous spray removal values and the values used for this current AST submittal are based on the differences in spray timing and when the appropriate DFs are reached, due to the assumed AST source-term release timing. The licensee's modeling of fission product removal in containment follows the guidance in RG 1.183, and is acceptable to the NRC staff.

Exelon's analysis assumed the containment leakage is at an increased L_a of 0.2 percent per day for the first 24 hours and 0.1 percent per day thereafter. This value is 2 times the current licensing basis containment leakage value, and supports the proposed change to TS 5.5.16, "Containment Leakage Rate Testing Program." Exelon used revised atmospheric dispersion factors for release from the containment in the LOCA radiological consequences analysis. The NRC staff's review of the revised atmospheric dispersion factors is discussed above in Section 3.1 of this safety evaluation (SE).

The NRC staff has reviewed the licensee's calculations, including description of the analysis inputs, assumptions, and methodology, as well as their calculational code input and output provided by letter dated December 9, 2005. The NRC staff finds that Exelon's modeling of the

LOCA containment leakage pathway follows the guidance of RG 1.183, and is, therefore, acceptable.

3.2.3 LOCA ECCS Leakage

Exelon calculated the dose due to leakage from ECCS components outside containment. RG 1.183, Appendix A, Section 5, states that the release source may also include leakage through valves isolating interfacing systems. See also Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," dated September 19, 1991. The dose contribution from leakage from the ECCS does not include back leakage into the RWST. By letter dated November 28, 2005, Exelon provided sufficient justification that no recirculation pathway exists to the RWST, and the NRC staff agrees that no dose calculation is necessary for potential back leakage to the RWST (as discussed in Section 3.2.1 of this SE). The ECCS leakage is assumed to start when ECCS recirculation initiates and continues at a constant rate until the end of the accident period at 30 days. The ECCS is assumed to leak 276,000 cc/hr, which is an increase to the previously assumed leakage value. This value is more conservative than RG 1.183 guidance of twice the value of maximum permitted recirculation loop leakage, which is 15,249 cc/hr for Braidwood and 13,294 cc/hr for Byron. The ECCS leakage value of 276,000 cc/hr was calculated by the licensee to be the upper bound that allows ECCS operability after 30 days without needing makeup water added. The source term for the ECCS leakage is the radioactivity in the containment sump water, which is assumed to consist of only the iodine isotopes. The iodine species in the ECCS leakage release are assumed to be 97 percent elemental iodine and 3 percent organic iodine compounds. The ECCS leakage flashing fraction was assumed to be 10 percent for the duration of the event. Exelon used revised atmospheric dispersion factors for release from the ECCS to the auxiliary building and out through the plant stack. The NRC staff's review of the revised atmospheric dispersion factors is discussed above in Section 3.1 of this safety evaluation.

The NRC staff has reviewed the licensee's calculations, including description of the analysis inputs, assumptions, and methodology, as well as their calculational code input and output provided by letter dated December 9, 2005. The NRC staff finds that Exelon's modeling of the LOCA ECCS leakage pathway follows the guidance of RG 1.183, and is, therefore, acceptable.

3.2.4 Control Room Modeling

The air volumes of the CR at Byron and Braidwood are 230,830 cubic feet (ft^3) and 232,872 ft^3, respectively. For its analysis, Exelon assumed a volume of 200,000 ft^3 to model the CR at either Byron or Braidwood. The assumed volume, although smaller than the actual air volume for either CR, is conservative for the calculation of dose in the CR. The assumed smaller volume, combined with high ventilation flow rates, is conservative in the analysis because the maximum activity concentration is reached more quickly than if the actual CR volume were used.

The licensee's CR dose analysis model of the filtered emergency intake included reduced CR ventilation filtration system makeup filter charcoal efficiency and increase in filter bypass, to account for changes to filter and system surveillance as proposed in changes to TS 5.5.11 b and c. The licensee assumed an effective makeup charcoal filter efficiency to approximate the increase in unfiltered flow to the CR by the assumed bypass of 1 percent of the makeup flow rate by the makeup filter. The licensee's assumed effective makeup charcoal filter efficiency of

95 percent is a reduction of 1 percent from the 96 percent efficiency that could be credited according to RG 1.52 for the proposed test penetration criterion. The NRC staff performed a sensitivity analysis for the LOCA containment release pathway analysis to evaluate the licensee's use of an effective filter efficiency instead of explicitly modeling the bypass by the makeup filter. The NRC staff's explicit model reduced the flow through the makeup filter by 1 percent of the flow and increased the filter efficiency to 96 percent, and modeled a separate unfiltered intake pathway for the 1 percent flow bypass. The dose results in the CR for this sensitivity analysis were the same for both cases, within the uncertainty of the calculation.

In the emergency mode of operation, the Byron and Braidwood CR filtration systems have a filtered recirculation flow of 43,500 cfm and filtered outside air make-up airflow of 6,000 cfm (± 10 percent), totaling 49,500 cfm of total combined flow to the CR volume. The licensee has considered failure scenarios which could lead to an additional 1,500 cfm of filtered intake into the CR through the unused train of the makeup system. Exelon rounded this additional 1,500 cfm up to 2,000 cfm, and added it to the total combined flow to assume a revised 51,500 cfm total combined flow to the control room in the emergency mode. A portion of this intake flow goes to the upper cable spreading room volume, and is un-recirculated. For Byron, this un-recirculated intake flow is 1,319 cfm, and for Braidwood, the un-recirculated intake flow is 2,430 cfm. The un-recirculated intake flow for each plant is subtracted from the 51,500 cfm total combined flow, making the assumed adjusted combined flow to the CR 50,181 cfm for Byron, and 49,070 cfm for Braidwood. The total make-up flow of 8,000 cfm (i.e., 6,000 cfm make up air and 2,000 cfm from the unused train failure) is adjusted for each plant, using the ratio of the adjusted intake flow (e.g., 50,181 cfm for Byron) to the total combined flow of 51,500 cfm. This results in filtration system emergency mode operation adjusted make-up flows of 7,795 cfm (± 10 percent) and 7,623 cfm (± 10 percent) to the Byron and Braidwood control rooms, respectively. Exelon assumed make-up filtration efficiencies of 99 percent for particulates and 95 percent for elemental and organic iodine. The assumed 95 percent charcoal filter efficiency is reduced compared to the value used in the previous analysis of record for Byron and Braidwood.

The licensee's analysis does not credit the CR filtration system emergency mode of operation for the first 30 minutes of the accident. During this time, intake of unfiltered make up air is assumed. The licensee stated that this assumption is intended to simulate an allowance for delayed initiation of the CR ventilation emergency mode of operation; however, no changes were proposed for the CR ventilation filtration system actuation instrumentation TS. For the first 30 minutes, the licensee did not consider potential leakage through the unused makeup train, as determined by sensitivity analyses. The additional flow would serve to clear out the radioactivity faster from the CR than if it were not considered in the analysis. Therefore, the NRC staff agrees that not including an additional 2,000 cfm of flow through the unused make-up train is conservative for the CR dose analysis. Without this additional flow, the adjusted unfiltered intake air flow during the first 30 minutes is calculated to be 5,840 cfm (± 10 percent) and 5,705 cfm (± 10 percent) to the Byron and Braidwood CRs, respectively. In the licensee's analysis, the unfiltered intake flow rate during this initial 30-minute period was assumed to be 6,424 cfm, which is the upper bound of the Byron 5,840 cfm makeup rate.

Exelon conservatively reduced the recirculation rate by using the - 10 percent lower bound value recirculation train flow rate of 39,150 cfm. The recirculation train filtration efficiencies are 80 percent for particulate and 90 percent for elemental and organic iodine. These are the current licensing basis values for CR recirculation filters.

In addition, 1,000 cfm of unfiltered inleakage is assumed for the duration of the event. This value bounds the maximum measured inleakage value of 68 standard cubic feet per minute (scfm) for Byron, and 29.3 scfm for Braidwood. The CR unfiltered inleakage was measured by means of tracer gas testing based on the American Society for Testing and Materials (ASTM) standard ASTM E-741-00, “Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.” The tests were completed on November 1, 2004, for Byron and November 7, 2004, for Braidwood. The NRC staff finds that the licensee’s assumption of 1,000 cfm CR unfiltered inleakage is conservative, compared to the actual measured values, and is acceptable.

Exelon calculated revised atmospheric dispersion factors for the CR, which were used in the CR analysis. The NRC staff’s review of the revised atmospheric dispersion factors is documented above in Section 3.1 of this SE.

3.4.2.1 Control Room Gamma Shine Dose

Exelon analyzed the gamma shine dose to the CR operators for the bounding LOCA accident. The contributors to CR doses due to gamma shine are identified in UFSAR Table 6.4-1 as being containment building airborne activity, the post-LOCA plume surrounding the CR, and radioactivity accumulated on the CR filters. The licensee’s assumed 200,000 ft^3 CR volume is larger than the 70,275 ft^3 actual CR shielded volume, so that the CR internal shine dose continues to be conservatively calculated. The licensee evaluated the containment activity shine dose and found that the current UFSAR containment activity shine dose contribution of 0.023 rem whole body is slightly conservative for AST conditions. The containment activity shine dose contribution was therefore taken to be 0.023 rem TEDE. For the external plume shine dose to the CR, because the shine dose is an external dose, the previously calculated whole-body dose can be considered to be approximately equivalent to the deep-dose equivalent used in calculating the TEDE. There is no internal dose component to the shine dose TEDE. The licensee adjusted the UFSAR current analysis small 0.003 rem whole-body external plume shine dose by multiplying the dose by a factor of 5 and converting the dose to the equivalent deep-dose equivalent, to yield 0.015 rem TEDE. This conservative multiplier accounts for the factor of 2 increase in primary containment leak rate assumptions, and other increased contributors such as ECCS iodine release. Additional calculational margin is also included by adjusting the current values because the revised containment release CR χ/Q is based on a diffuse area source lower than the values historically used in the existing dose analysis. The licensee reanalyzed the fission product filter loading using AST assumptions for the LOCA, and compared the results with those that would be determined using RG 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” assumptions. The current UFSAR analysis and pre-AST source terms were found to be bounding by the licensee. The NRC staff has reviewed the licensee’s analysis and finds that it follows the guidance in RG 1.183 for the AST and does show that the current UFSAR analysis remains bounding. Therefore, the licensee converted the existing 0.013 rem whole-body dose to 0.013 rem TEDE dose.

The licensee calculated total gamma shine dose in the CR is 0.05 rem TEDE. The NRC staff reviewed the licensee’s description of its evaluation of the CR shine dose and the supporting analyses, and finds that the licensee used assumptions that follow RG 1.183 guidance for source terms and release and are appropriate for calculating the shine dose in the CR.

3.2.5 Technical Support Center Dose

The Byron and Braidwood UFSAR, Appendix E, "Requirements Resulting from [Three Mile Island, Unit 2 (TMI-2)] Accident," addresses requirements resulting from the Three Mile Island, Unit 2 accident, discussed in NUREG-0737, "Clarification of TMI Action Plan Requirements." Exelon determined that the dose in the technical support center (TSC) is impacted by the implementation of an AST and the associated plant changes. Exelon recalculated the TSC dose using the same AST LOCA release assumptions as discussed above for the CR and offsite doses, but using TSC-specific ventilation system parameters and atmospheric dispersion factors. As in the CR analysis, the licensee did not take credit for operation of the emergency filtration system in the TSC for 30 minutes. The TSC nominal intake rate of 900 cfm was increased by 10 percent during the inital 30-minute period to allow for uncertainty. For the duration of the event, a TSC intake rate of 810 cfm (900 cfm - 10 percent) was determined to be conservative. Unfiltered inleakage was assumed to be 450 cfm (half of the nominal intake rate). The intake filter efficiency is 95 percent for elemental and organic iodine and 99 percent for particulates. The TSC recirculation flow is assumed to be 990 cfm, which is the 1,100 cfm nominal flow minus 10 percent to allow for uncertainty and minimize activity removal. No credit is taken for recirculation filtration for the first 30 minutes. Exelon used TSC-specific χ/Qs that were calculated using RG 1.194 methodology. The reanalyzed TSC dose results are within 5 rem TEDE, and are, therefore, acceptable to the NRC staff.

3.2.6 LOCA Results

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements and the Byron and Braidwood current licensing basis and, also reviewed the RADTRAD calculation output provided by the licensee. Exelon's LOCA dose analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 4. The licensee's limiting calculated LOCA dose results are given in Table 3. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for a LOCA. These TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.3 FHA Radiological Consequences Analysis

Exelon analyzed the radiological consequences of a FHA, which is defined as the drop of a spent fuel assembly onto the spent fuel pool floor or the core, resulting in the postulated rupture of the cladding of all the fuel rods in one assembly. The licensee considered two accident locations; the FHB and the containment. Exelon proposed changes to the applicability requirements during movement of irradiated fuel assemblies, consistent with TSTF-51, Revision 2, which was approved by the NRC by letter dated November 1, 1999 (ADAMS Accession Number ML993190284). TSTF-51 allows changes to TS operability requirements for certain engineered safety features such that they are not required to be operable for fuel handling or core alterations after sufficient radioactive decay has occurred to ensure that doses remain within limits. Exelon has performed analyses of the FHA assuming that the dropped assembly has undergone 48 hours of radioactive decay, in order to support the proposed containment and ventilation systems operability relaxations. For fuel movement before 48 hours after shutdown, Exelon also performed an analysis assuming six hours of decay. Exelon states that movement of recently irradiated fuel in less than 6 hours is not considered to

be possible because of the activities that must be performed so that fuel movement in the vessel can take place. These analyses support the licensee's proposed TS definition of "recently irradiated fuel" as fuel that has occupied part of a critical reactor core within the previous 48 hours.

Exelon's FHA analyses followed the guidance of RG 1.183. The analyses assumed a bounding core isotopic inventory based on a power level of 3,658.3 MWt, which is 102 percent of the rated thermal power to account for measurement uncertainty. For the FHA and non-LOCA transients and accidents, the fuel/clad gap fission product inventories in Table 3 of RG 1.183 were increased by a factor of 2. The factor of 2 was used to offset the fact that some fuel assemblies would exceed rod power/burnup criteria in RG 1.183. The factor of 2 was applied to all assemblies.

For the FHA, all of the fuel rods in the limiting assembly are assumed to fail, releasing their fuel/clad gap fission-product inventory. Increasing the values in Table 3 by a factor of 2 for all of the fuel rods in the assembly is conservative based upon less than 50 percent of these rods are expected to exceed the specified applicability criteria in RG 1.183. Further, the Advance Nodal Computer (ANC) calculations provided by the licensee indicate that rod power does not exceed the applicability criteria for rod power by more the 10 percent. In addition, the cycle-maximum radial peaking factor is conservatively applied to the fuel rods in the limiting, high burnup assembly which creates a conservative composite worst-case fuel configuration (high power with high gap inventory). A radial peaking factor of 1.7 was also applied to the spent fuel source term. Following the guidance in RG 1.183, an effective iodine DF of 200 was used to account for iodine removal in the 23-foot water depth above the damaged fuel.

The CR modeling for the FHA analysis was the same as was described above for the LOCA analysis. Exelon used revised atmospheric dispersion factors in the radiological consequences analysis. The NRC staff's review of the revised atmospheric dispersion factors is discussed above in Section 3.1 of this SE. The FHA analysis is performed using the most conservative values for assumptions and inputs applicable to either of the units for Byron and Braidwood, and therefore, the results are conservative for all four units.

For the case where the FHA occurs at 48 hours after shutdown, no filtration of the release or automatic isolation of the accident location (either the containment or the FHB) is assumed. Essentially all of the radioactivity released from the damaged fuel is assumed to reach the environment within 2 hours. The release point for the FHA in the FHB is the plant vent, which has the highest χ/Q for the major openings in the FHB. The release point for the FHA in the containment is the plant vent. A release through the personnel/equipment hatch from the containment to the auxiliary building would be exhausted through the auxiliary building ventilation system to the plant vent, with the same χ/Q values as for the containment purge or FHB releases.

For the case where the FHA occurs at 6 hours after shutdown, containment closure is established and/or the FHB ventilation system (as applicable to the accident location) and the CR filtration system are operable according to the TS requirements. Exelon's analysis of the FHA for recently irradiated fuel is described in detail in calculation number BYR04-047 & BRW-04-0041-M, Revision No. 1, "Re-analysis of Fuel Handling Accident (FHA) Using Alternative Source Terms," which is Attachment 3 to the licensee's letter dated December 9,

2005. The licensee assumed that the FHB ventilation envelope is isolated during the event with the FHB ventilation system operable, and the FHB ventilation system filtration efficiencies of 99 percent for particulates and 90 percent for elemental and organic iodine. The dose results for the FHA at 6 hours post shutdown, with filtration by the FHB ventilation system, are 5.3 rem at the EAB for the worst 2 hours, 0.44 rem at the LPZ for the duration of the accident and 1.56 rem in the CR for the duration of the accident. These calculated doses are within the SRP 15.0.1 regulatory acceptance criteria for the FHA.

The FHA dose analysis results may be used to justify a future design change to allow for the FHB or containment not being isolated with a hatch left open during core alterations or movement of the appropriate fuel, provided that the capability for closure is maintained. In its letter dated February 13, 2006, Exelon described how Byron and Braidwood would mitigate the consequences of the FHA in accordance with TSTF-51 and consistent with the guidance in NUMARC 93-01, “Safety Assessment for Removal of Equipment from Service During Shutdown Conditions,” Revision 3. The current licensing basis for Byron and Braidwood permits the equipment hatch to remain open during movement of fuel in the containment. The equipment hatches at all four units open into the FHB, and do not open directly to the outside environment. When the equipment hatch is open, the associated unit containment becomes part of the FHB ventilation envelope. The FHA ventilation system is designed to mitigate the consequences of an FHA by filtration before release through the plant vent. In Attachment 5 to its February 15, 2005, submittal, Exelon made commitments to mitigate the consequences of a potential FHA. Exelon committed to have at least one train of the FHB ventilation system available when moving irradiated fuel in the containment or FHB, close any openings from the containment or FHB to the outside environment and establish air flow in the proper direction within 1 hour, and monitor the containment and FHB effluent consistent with the Byron and Braidwood TS 5.5.4, Radioactive Effluent Controls Program. By these actions, the licensee would provide for control and filtration of any potential release after 1 hour, which would lower the calculated dose compared to the analysis, which assumed no filtration, but still meets the regulatory dose acceptance criteria for fuel that has decayed 48 hours.

The NRC staff reviewed the information provided in the licensee’s AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon’s FHA analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 5. The licensee’s calculated dose results are given in Table 3. The licensee’s calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA. These TEDE criteria are 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.4 CREA Radiological Consequences Analysis

Exelon's dose analysis of a CREA considered two separate scenarios for fission product release to the environment; (1) primary-to-secondary leakage and secondary steaming through the SG PORVs until cold shutdown, or (2) a breach of the RPV and containment leakage for 30 days. The doses from the two release scenarios are not added together, but are considered separately. Exelon used revised atmospheric dispersion factors in the radiological consequences analysis. The NRC staff's review of the revised atmospheric dispersion factors is discussed above in Section 3.1 of this SE.

For the source term, the licensee's dose analysis assumed 10 percent of the core fuel rods are damaged and immediately release the fission products within the gap to the RCS coolant. The damaged fuel is assumed to have operated at a radial peaking factor of 1.7. The fission product gap release fractions conform to RG 1.183 guidance of 10 percent of the core inventory of iodine and noble gas. Exelon's analysis assumed two times the RG 1.183 gap fractions of cesium and rubidium are released to account for possible damage to high-burnup fuel with a high linear heat generation rate. The analysis also assumed that 2.5 percent of the damaged rods immediately release melted fuel activity (i.e., 0.25 percent of the total core melts). The melted fuel release fractions are 100 percent noble gases, 50 percent of the iodines and 50 percent of the cesium and rubidium, which is in accordance with or bounds the guidance in RG 1.183.

For the containment leakage scenario, Exelon assumed that all the activity available for release was released immediately into the containment through the break. The containment leakage release pathway model assumed that the containment was leaking at the proposed TS containment L_a value of 0.2 volume percent per day for the first 24 hours, then leaks at half that value until the end of the accident, which was assumed to be 30 days. The licensee did not assume that iodine in the containment was removed through containment spray operation. Natural deposition of particulate iodine was assumed at the lower bound (10 percentile) value of the Powers algorithm based on NUREG/CR-6189 as in the LOCA analysis, which is in accordance with guidance in RG 1.183.

For the secondary steaming scenario, RCS integrity is maintained and radioactive materials in the reactor coolant are transported to the secondary coolant through primary-to-secondary leakage of 1 gallon per minute (gpm) total leakage. Activity in the secondary coolant at the TS limit of 0.1 μCi/gm Dose Equivalent I-131 is mixed with the released fuel activity leaked into the secondary coolant. The total activity in the secondary coolant is then assumed to be released to the environment through secondary coolant steaming through the SG PORVs until the plant is cooled down so that the primary and secondary systems are in equilibrium. The licensee calculated an average rate of release for use in the dose analysis, which was derived from calculated total time incremented mass releases. The licensee assumed an iodine partitioning factor of 0.01 in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released. The licensee assumed that some cesium and rubidium are released through the secondary coolant with a partitioning factor of 0.0055, which bounds the value of 0.00529 shown in ANSI standard ANS/ANSI 18.1 -1999, "Radioactive Source Term for Normal Operation for Light Water Reactors," for Cs-134, which has the largest partition factor of these isotopes. The NRC staff finds that the licensee has determined a reasonable value for the cesium and rubidium partitioning factor.

Releases from the SG PORVs were considered by the licensee to be elevated releases due to the high steaming rates, therefore the CR atmospheric dispersion factors were reduced by a factor of 5 in accordance with regulatory guidance. The NRC staff's review of this reduction in the χ/Qs for the SG PORV release is discussed above in Section 3.1 of this SE. The CR modeling was the same as was described above for the LOCA analysis.

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon's CREA analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 6. The licensee's calculated dose results are given in Table 3. The licensee's calculated doses from either release scenario are within the SRP 15.0.1 radiological dose acceptance criteria for a CREA. These TEDE criteria are 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.5 LRA Radiological Consequences Analysis

Exelon's dose analysis of the reactor coolant pump (RCP) LRA assumed that an RCP becomes inoperable and loss of primary coolant circulation may result in as much as 2 percent of the core fuel rods experiencing subsequent cladding damage. The fission products in the gap of those damaged fuel rods were assumed released immediately into the primary coolant. Exelon used the gap fractions given in RG 1.183 Table 3 that would be applicable to the LRA, with the fractions for cesium and rubidium doubled as described above for the CREA which is conservative because on a core-wide basis, only a small fraction of the fuel rods exceed the applicability criteria in RG 1.183. Further, during core-wide transients, the high power, low burnup rods (containing less gap inventory) are more prone to clad failure than the low power, high burnup rods (containing more gap inventory). In addition, the licensee stated that the cycle-maximum radial peaking factor was applied to all failed fuel rods. The fuel was assumed to have been operating at a radial peaking factor of 1.7. No fuel melting is calculated to occur for the LRA.

Exelon analyzed two cases for the LRA. The first case is modeled to calculate the doses due to the activity that was instantaneously released into the RCS from the postulated damaged fuel, assuming the primary coolant activity is elevated in a pre-accident 60 μCi/gm Dose Equivalent I-131 iodine spike. The second case models the additional dose from the secondary coolant activity at the TS limit of 0.1 μCi/gm Dose Equivalent I-131, instead of the primary coolant activity. The dose results of each case are added together to give a total LRA dose to compare to the regulatory acceptance criterion.

The radioactive materials in reactor coolant are transported to the secondary coolant through a total primary-to-secondary leakage of 1 gpm. For the first LRA case, primary coolant activity of 60 μCi/gm Dose Equivalent I-131 is assumed to mix with failed fuel activity in the RCS. For the second LRA case, activity in the secondary coolant of 0.1 μCi/gm Dose Equivalent I-131 is mixed with the failed fuel activity leaked through the SG tubes. The total activity in the secondary coolant is then assumed to be released to the environment through secondary coolant steaming through the SG PORVs until the plant is cooled down so that the primary and secondary systems are in equilibrium at 40 hours. Exelon calculated an average cumulative mass release rate through the SG PORVs, which was derived from calculated total time

incremented mass releases. Exelon also assumed failure of one SG PORV in the open position for 20 minutes to conservatively maximize the activity release. The licensee assumed an iodine partitioning factor of 0.01 in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released. The licensee assumed that some cesium and rubidium are released through the secondary coolant with a partitioning factor of 0.0055, which bounds the value of 0.00529 shown in ANSI standard ANS/ANSI 18.1 -1999 for Cs-134, which has the largest partition factor of these isotopes. The NRC staff finds that the licensee has determined a reasonable value for the cesium and rubidium partitioning factor, especially in the light that RG 1.183 guidance does not model release of those isotopes through the secondary coolant pathway.

Releases from the SG PORVs were considered by the licensee to be elevated releases due to the high steaming rates, therefore the CR atmospheric dispersion factors were reduced by a factor of 5 in accordance with regulatory guidance. The NRC staff's review of this reduction in the χ/Qs for the SG PORV release is discussed above in Section 3.1 of this SE. The CR modeling was the same as was described above for the LOCA analysis.

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon's LRA analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 7. The licensee's calculated dose results are given in Table 3. The licensee's calculated doses from either release scenario are within the SRP 15.0.1 radiological dose acceptance criteria for a LRA. These TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.6 MSLB Radiological Consequences Analysis

The MSLB accident is postulated as a break of one of the large steam lines outside of containment. For the three intact SG loops, primary-to-secondary coolant leakage transfers activity into the secondary coolant, where it is available for release to the environment through steaming through the SG PORVs. For the broken steam line, the primary-to-secondary coolant leakage is assumed to be released directly from the RCS to the environment because of "dry-out" conditions in the faulted SG.

No fuel failure was predicted for the MSLB, so the radiological source term was based on the primary coolant activity. In accordance with RG 1.183 guidance, two iodine spiking cases were considered (1) pre-accident elevated iodine concentration equal to the TS maximum iodine concentration of 60 μCi/gm Dose Equivalent-131, and (2) accident-initiated iodine spiking of 500 times the iodine appearance rate that equates to the TS equilibrium iodine concentration limit of 1 μCi/gm Dose Equivalent I-131. The results of each case were compared to the appropriate regulatory acceptance criterion.

The radioactive materials in reactor coolant are released to the environment through primary-to-secondary leakage of 0.5 gpm to the faulted SG and 0.218 gpm per unaffected SG, and then released through the break in the main steam line until isolation and SG secondary coolant steaming through the unaffected SG PORVs until the plant is cooled down so that the primary and secondary coolant are at equilibrium at 40 hours. Exelon calculated an average

cumulative mass release rate through the SG PORVs, which was derived from calculated total time incremented mass releases. Exelon also assumed failure of one SG PORV in the open position for 20 minutes to conservatively maximize the activity release. The licensee assumed an iodine partitioning factor of 0.01 in the unaffected SGs in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released.

Releases from the SG PORVs were considered by the licensee to be elevated releases due to the high steaming rates, therefore the CR atmospheric dispersion factors were reduced by a factor of 5 in accordance with regulatory guidance. The NRC staff's review of this reduction in the χ/Qs for the SG PORV release is discussed above in Section 3.1 of this SE. The CR modeling was the same as was described above for the LOCA analysis.

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon's MSLB analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 8 of the licensee's February 15, 2005 submittal. The licensee's calculated doses from both analyzed iodine spiking cases are within the SRP 15.0.1 radiological dose acceptance criteria for an MSLB without fuel damage. For the pre-accident iodine spiking case, these TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident. For the accident-induced iodine spiking case, these TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.7 SGTR Radiological Consequences Analysis

The SGTR accident is postulated as a complete severance of a single SG tube. Exelon's SGTR radiological consequences analysis evaluated two separate radiological source term cases; an accident initiated iodine spike case and a pre-accident iodine spike case. The accident initiated iodine spike case assumed the primary coolant activity was initially at the long-term TS limit of 1.0 μCi/gm Dose Equivalent I-131 when the event occurs. The accident was assumed to cause the iodine concentration to spike by addition of iodine activity by a factor of 335 times the equilibrium iodine appearance rate for 8 hours. The pre-accident iodine spike case assumed the primary coolant iodine concentration was at an increased value of 60 μCi/gm Dose Equivalent I-131. This is the TS limit for full power operation following an iodine spike for up to 48 hours. The results of each case were compared to the appropriate regulatory acceptance criterion.

For both cases, all modeling other than the activity source term is the same. The primary-to-secondary coolant leakage of 0.218 gpm per SG goes to the three intact SGs. The activity in the coolant is available for release to the environment through secondary coolant steaming through the SG PORVs. The licensee assumed an iodine partitioning factor of 0.01 in the intact SGs in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released. Steaming from the intact SG is modeled from the time of RCS cooldown initiation until the primary and secondary coolant are in equilibrium at 3,500 seconds.

Primary coolant was assumed to pass through the ruptured SG tubes and be available for release to the outside environment by steaming through the ruptured SG PORV. A portion of

the rupture flow flashes directly to steam, while the remainder mixes with the secondary coolant for subsequent steaming through the PORV. The flashing fraction and partitioning fractions for the ruptured SG are in accordance with RG 1.183 guidance.

The licensee also modeled the full power steam activity released through the condenser until the time of the reactor trip and loss of offsite power. Exelon calculated a mass release rate of 16,040,000 pounds-mass per hour for the initial 445 seconds for this release pathway. The iodine partitioning factor of 0.01 is the same as for the release through the SG PORVs, and partitioning in the condenser is also considered.

Releases from the SG PORVs were considered by the licensee to be elevated releases due to the high steaming rates, therefore the CR atmospheric dispersion factors were reduced by a factor of 5 in accordance with regulatory guidance. The NRC staff's review of this reduction in the χ/Qs for the SG PORV release is discussed above in Section 3.1 of this SE. The CR modeling was the same as was described above for the LOCA analysis.

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon's SGTR analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 9. The licensee's calculated doses from both analyzed iodine spiking cases are within the SRP 15.0.1 radiological dose acceptance criteria for an SGTR without fuel damage. For the pre-accident iodine spiking case, these TEDE criteria are 25 rem at the EAB for the worst 2 hours, 25 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident. For the accident-induced iodine spiking case, these TEDE criteria are 2.5 rem at the EAB for the worst 2 hours, 2.5 rem at the LPZ for the duration of the accident, and 5 rem in the CR for the duration of the accident.

3.8 Environmental Qualification

The licensee has elected for Byron and Braidwood to retain the TID-14844 assumptions for performing the required environmental qualification (EQ) analyses.

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, the integrated doses calculated with the AST exceeded those calculated with TID-14844 after 42 days for a pressurized-water reactor, because of the 30 percent versus 1 percent release of cesium according to NUREG-1465. The licensee EQ program for Byron and Braidwood is based on a post-accident period of 30 days. The continued use of the TID-14844 source term provides integrated doses for equipment which envelope those that would be calculated using AST. Therefore, following implementation of AST, Byron and Braidwood will continue to meet their commitment to 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," 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.

3.9 Technical Specification changes

TSTF-51 allows changes to TS operability requirements for certain engineered safety features such that they are not required to be operable for fuel handling or core alterations after sufficient radioactive decay has occurred to ensure that doses remain within limits. TSTF-51 states "To support this change in requirements during the handling of irradiated fuel, the OPERABILITY requirements during CORE ALTERATIONS for ESF mitigation features are deleted. The accidents postulated to occur during core alterations, in addition to fuel handling accidents, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Since the only accident postulated to occur during CORE ALTERATIONS that results in a significant radioactive release is the fuel handling accident, the proposed Technical Specification requirements omitting CORE ALTERATIONS is justified." Exelon has performed analyses of the FHA assuming that the dropped assembly has undergone 48 hours of radioactive decay, in order to support the proposed containment and ventilation systems operability relaxations. The NRC staff reviewed the information provided in the licensee's AST submittal and supplements, the Byron and Braidwood current licensing basis, and the RADTRAD calculation output provided by the licensee. Exelon's FHA analysis used assumptions and inputs that follow the guidance in RG 1.183. The licensee's calculated doses are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA.

3.9.1 TS Section 1.1, "Definitions," DOSE EQUIVALENT I-131

Exelon proposed to revise the definition of DOSE EQUIVALENT I-131 in TS Section 1.1 to remove the word "thyroid" and add a reference to FGR-11, Table 2.1.

The existing definition is based on the thyroid dose conversion factors provided in specified tables in either Atomic Energy Commission (AEC) TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," RG 1.109, Revision1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," or International Commission on Radiation Protection Publication 30 (ICRP 30), "Limits for Intakes of Radionuclides by Workers." The proposed change to the definition of Dose Equivalent I-131 conforms to the implementation of the AST and the TEDE criteria in 10 CFR 50.67. The new citations are as cited in RG 1.183 and, are therefore, acceptable. The FGR-11 dose conversion factors were used by Exelon in the revised analyses of the CREA, LRA, MSLB, and SGTR accidents, which were found acceptable by the NRC staff as discussed above. The NRC staff finds the proposed change to the TS 1.1 definition DOSE EQUIVALENT I-131 acceptable.

3.9.2 TS Section 1.1, "Definitions" RECENTLY IRRADIATED FUEL

The proposed change adds the definition of RECENTLY IRRADIATED FUEL. RECENTLY IRRADIATED FUEL is defined as, "Fuel that has occupied part of a critical reactor core within the previous 48 hours. Note that all fuel that has been in a crititcal core is referred to as irradiated fuel."

The licensee's revised FHA dose analysis to support relaxed requirements for containment and ventilation systems consistent with TSTF-51 included an isotopic source term that assumed that 48 hours of radioactive decay has elapsed prior to the accident. The FHA analysis was found acceptable by the NRC staff as discussed above. The licensee's FHA dose analysis assumptions, inputs and results support the proposed TS bases definition of RECENTLY IRRADIATED FUEL as fuel that has occupied part of a critical reactor core within the previous 48 hours and, is therefore acceptable to the NRC staff.

3.9.3 TS 3.3.7, "VC Filtration System Actuation Instrumentation"

Required Action E.1, "Suspend CORE ALTERATIONS" and its associated completion time are deleted. The NRC staff reviewed the proposed changes to TS 3.3.7 and found they are consistent with TSTF-51 and are supported by the licensee's AST analysis. As mentioned in Section 3.3 of this SE, the NRC staff has previously endorsed TSTF-51, therefore the NRC staff finds the changes to TS 3.3.7 acceptable.

3.9.4 TS 3.3.8, "FHB Ventilation System Actuation Instrumentation"

The words "irradiated fuel" in Required Actions B.2.1 and B.2.2 are replaced with the defined term RECENTLY IRRADIATED FUEL. Required Action B.2.3 to suspend CORE ALTERATIONS with the equipment hatch not intact is deleted. As discussed in Section 3.9.2 of this SE, the change from "irradiated fuel" to "RECENTLY IRRADIATED FUEL" is supported by the licensee's AST analysis, and the use of "RECENTLY IRRADIATED FUEL," in TS 3.3.8 is consistent with TSTF-51. The deletion of Required Action B.2.3 is consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.3.8 acceptable.

3.9.5 TS Table 3.3.8-1, "FHB Ventilation System Actuation Instrumentation"

In Notes (a) and (b), the term "irradiated fuel" is replaced with the defined term RECENTLY IRRADIATED FUEL. Note c), applicable to Function 1, "Fuel Handling Building Radiation" is deleted, as well as the Note itself. Deletion of the note eliminates the requirement to have the FHB radiation monitors operable during CORE ALTERATIONS with the equipment hatch not intact. As discussed in Section 3.9.2 of this SE the change from "irradiated fuel" to "RECENTLY IRRADIATED FUEL," is supported by the licensee's AST analysis, and the use of "RECENTLY IRRADIATED FUEL," in TS Table 3.3.8-1 is consistent with TSTF-51 and supported by the licensee's AST analysis. The deletion of Required Action Note c) is consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.3.8 acceptable.

3.9.6 TS 3.7.10, "VC Filtration System"

Required Action C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions C.2.2 and C.2.3 are renumbered to support the deletion of Required Action C.2.1.

Required Action D.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions D.2 and D.3 are renumbered to support the deletion of Required Action D.1.

The deletion of Required Actions C.2.1 and D.1 are consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.7.10 acceptable.

3.9.7 TS 3.7.11, "VC Temperature Control System"

Required Action C.2.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions C.2.2 and C.2.3 are renumbered to support the deletion of Required Action C.2.1.

Required Action D.1 and its completion time regarding suspension of CORE ALTERATIONS are deleted. Required Actions D.2 and D.3 are renumbered to support the deletion of Required Action D.1.

The deletion of Required Actions C.2.1 and D.1 are consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.7.11 acceptable.

3.9.8 TS 3.7.13, "FHB Ventilation System"

The term "irradiated fuel" in the applicability statement is replaced with the words "RECENTLY IRRADIATED FUEL." The reference to suspension of CORE ALTERATIONS is removed.

The term "irradiated fuel" in Required Actions B.2.1 and B.2.2 is replaced with the term RECENTLY IRRADIATED FUEL. Required Action B.2.3 regarding the suspension of CORE ALTERATIONS is deleted.

The term "irradiated fuel" is replaced with the words "RECENTLY IRRADIATED FUEL" in Required Actions C.1 and C.2. Required Action C.3 regarding the suspension of CORE ALTERATIONS is deleted.

In the NOTE to SR 3.7.13.3, the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL, and the words "or CORE ALTERATIONS" are deleted.

In the NOTE to SR 3.7.13.5, the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL.

As discussed in Section 3.9.2 of this SE, the change from "irradiated fuel" to "RECENTLY IRRADIATED FUEL" is supported by the licensee's AST analysis, and the use of "RECENTLY IRRADIATED FUEL," in TS 3.7.12 is consistent with TSTF-51. The deletion of the requirement

to suspend CORE ALTERATIONS is consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.7.13 acceptable.

3.9.9 TS 3.9.4, "Containment Penetrations"

The licensee requested a change to the Note to LCO 3.9.4 as follows: the reference to "LCO" is change to "TS". The licensee stated that this is an editorial change.

In response to the NRC staff's request for additional information, by letter dated July 14, 2006, the licensee stated that the Note is being revised from:

Item a. only required when the Fuel Handling Building Exhaust Filter Plenum Ventilation System is not in compliance with TS 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System."

To the following:

LCO 3.9.4.a is not required to be met when in compliance with LCO 3.7.13, "Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System," or its associated Conditions and Required Actions.

The NRC staff reviewed this requested change and agrees with the licensee that it is an editorial change, and therefore, is acceptable.

The term "irradiated fuel" in the applicability statement is replaced with the words "RECENTLY IRRADIATED FUEL." The reference to the suspension of CORE ALTERATIONS is deleted.

In TS 3.9.4, Required Action A.1, the reference to the suspension of CORE ALTERATIONS and its associated completion time are deleted. Required Action A.2 is renumbered and the term "irradiated fuel" is replaced with the term RECENTLY IRRADIATED FUEL.

As discussed in Section 3.9.2 of this SE the change from "irradiated fuel" to "RECENTLY IRRADIATED FUEL" is supported by the licensee's AST analysis, and the use of "RECENTLY IRRADIATED FUEL" in TS 3.9.4 is consistent with TSTF-51. The deletion of the requirement to suspend CORE ALTERATIONS is consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.9.4 acceptable.

3.9.10 TS 3.9.7, "Refueling Cavity Water Level"

The reference in the applicability statement to the suspension of CORE ALTERATIONS is removed; however, the reference to "irradiated fuel assemblies" is retained in this location.

In TS 3.9.7, Required Action A.1, the reference to the suspension of CORE ALTERATIONS and its associated completion time, are deleted. Required Action A.2 is renumbered to Required Action A.1.

The deletion of the requirement to suspend CORE ALTERATIONS in TS 3.9.7 is consistent with TSTF-51 and supported by the licensee's AST analysis. Therefore, the NRC staff finds the changes to TS 3.9.7 acceptable.

3.9.11 TS 5.5.11, "Ventilation Filter Testing Program (VFTP)"

The licensee requested changes to TS 5.5.11 Items b and c. In Item b, the charcoal adsorber bypass acceptance criterion is changed from 0.05 percent to 1 percent. In Item c, the charcoal adsorber penetration acceptance criterion is changed from 0.5 percent to 2 percent. According to the licensee, these changes can be reflected by a reduction in assumed filtration efficiency from 99 percent to 95 percent, which was used in the AST dose analysis and maintains a safety factory of 2 for the penetration acceptance criterion.

The licensee's CR dose analysis model of the filtered emergency intake included reduced CR ventilation filtration system makeup filter charcoal efficiency and increase in filter bypass, to account for changes to filter and system surveillance as proposed in changes to TS 5.5.11 Items b and c. The licensee assumed an effective makeup charcoal filter efficiency to approximate the increase in unfiltered flow to the CR by the assumed bypass of 1 percent of the makeup flow rate by the makeup filter. The licensee's assumed effective makeup charcoal filter efficiency of 95 percent is a reduction of 1 percent from the 96 percent efficiency that could be credited according to RG 1.52 for the proposed test penetration criterion. The NRC staff performed a sensitivity analysis for the LOCA containment release pathway analysis to evaluate the licensee's use of an effective filter efficiency instead of explicitly modeling the bypass by the makeup filter. The NRC staff's explicit model reduced the flow through the makeup filter by 1 percent of the flow and increased the filter efficiency to 96 percent, and modeled a separate unfiltered intake pathway for the 1 percent flow bypass. The dose results in the CR for this sensitivity analysis were the same for both cases, within the uncertainty of the calculation. Therefore, the NRC staff finds these changes acceptable.

3.9.12 TS 5.5.16, "Containment Leakage Rate Testing Program"

The maximum allowable containment leakage rate leakage limit (L_a) is changed from 0.1 percent to 0.2 percent of containment air weight per day.

The licensee's LOCA containment leakage pathway dose analysis assumed that the total leakage from the containment is 0.2 percent of containment air weight per day for the first 24 hours, and half that amount (0.1 percent/day) for the remainder of the 30-day duration. These assumptions are in accordance with RG 1.183 guidance for PWR reactor LOCAs. The increased maximum allowable leakage rate of 0.2 percent containment air weight per day was used by Exelon in the revised analyses of the LOCA, which was found acceptable by the NRC staff as discussed above. The licensee's LOCA dose analysis assumptions, inputs and results support the proposed changes to TS 5.5.16, maximum allowable L_a at design accident pressure increase to 0.2 percent of containment air weight per day and, is therefore, acceptable to the NRC staff.

3.10 Withdrawn TS 3.7.10, "VC Filtration System," SR 3.7.10.4

The licensee in its February 15, 2005, submittal requested a change to SR 3.7.10.4. The licensee had proposed that the words "the upper cable spreading room at positive pressure of

20.02 inches water gauge and" be removed. The word "area" (i.e., in the statement, ". . . adjacent to the control room *area* during . . . ") would have been replaced with the word "envelope." The licensee in its March 17, 2006, supplement withdrew the proposed TS 3.7.10 change.

3.11 Technical Evaluation Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Exelon to assess the radiological impacts of implementation of an AST and changes to TS for containment leakage, ECCS leakage and control room filtration systems at Byron and Braidwood. The staff finds that Exelon 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 Exelon 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 control room doses will continue to comply with these criteria. Therefore, the proposed TS changes are acceptable with regard to the radiological consequences of postulated design basis accidents.

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

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (70 FR 24650; May 10, 2005). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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TABLE 1

Byron Units 1 & 2 and Braidwood Units 1 & 2
Control Room Atmospheric Dispersion Factors*
(χ/Q values in sec/m^3)

A. All Accidents Except for LOCA ECCS Leakage

Release Pathway	Accidents	Normal Intake	Emergency Intake				
		0-0.5 hr	0.5–2 hr	2–8 hr	8–24 hr	1–4 day	4–30 day
Containment Leakage	LOCA CREA[j]	1.73×10^{-3}	1.01×10^{-3}	7.25×10^{-4}	3.07×10^{-4}	2.07×10^{-4}	1.46×10^{-4}
Plant Vent	FHA	2.22×10^{-3}	2.46×10^{-3}	1.92×10^{-3}	8.14×10^{-4}	5.52×10^{-4}	4.40×10^{-4}
SG PORVs & Safety Valves	CREA[k] LRA MSLB[l] SGTR	1.77×10^{-3}	8.14×10^{-4}	6.98×10^{-4}	3.12×10^{-4}	1.95×10^{-4}	1.67×10^{-4}
MSLB	MSLB[m]	3.20×10^{-3}	1.70×10^{-2}	1.46×10^{-2}	6.68×10^{-3}	4.48×10^{-3}	3.31×10^{-3}

B. LOCA ECCS Leakage[n]

Release Pathway	Normal Intake	Emergency Intake					
	0–0.5 hr	0.5–1.8 hr	1.8–3.3 hr	3.3–8 hr	8–24 hr	1–4 day	4–30 day
Plant Vent	2.22×10^{-3}	1.92×10^{-3}	2.46×10^{-3}	1.92×10^{-3}	8.14×10^{-4}	5.52×10^{-4}	4.40×10^{-4}

* For each time interval, the licensee applied the worst case χ/Q value to both sites irrespective of the site for which it was actually calculated.

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[j]Used for CREA Case 1 where the ejected control rod is assumed to breach the RPV.

[k]Used for CREA Case 2 where RCS integrity is maintained and all activity is assumed to leak to the secondary side through the SG tubes.

[l]Intact SG loop releases.

[m]Broken steam line releases.

[n]For modeling LOCA ECCS leakage, the licensee applied the worst (highest) emergency intake χ/Q value (i.e., the 0 to 2-hour plant vent χ/Q value of 2.46×10^{-3} sec/m^3) at a point in time (1.8 hours into the event) which would maximize CR doses. The remaining time period within the 0.5 to 8-hour interval was modeled using the 2 to 8-hour plant vent χ/Q value of 1.92×10^{-3} sec/m^3.

TABLE 2

Braidwood Units 1 & 2 and Byron Units 1 & 2
Offsite Atmospheric Dispersion Factors*
(χ/Q values in sec/m^3)

Receptor	Accident	0–2 hr	2–8 hr	8–24 hr	1–4 day	4–30 day
EAB	CREA FHA LOCA LRA MSLB SGTR	5.36×10^{-4}	--	--	--	--
LPZ	CREA LRA MSLB SGTR	9.32×10^{-5}	4.50×10^{-5}	3.12×10^{-5}	1.41×10^{-5}	4.54×10^{-6}
	FHA LOCA	4.50×10^{-5}	4.50×10^{-5}	3.12×10^{-5}	1.41×10^{-5}	4.54×10^{-6}

* For each time interval, the licensee applied the worst case χ/Q value to both sites irrespective of the site for which it was actually calculated.

TABLE 3

Licensee Calculated DBA Radiological Consequences

TEDE (rem)						
Accident	EAB		LPZ		Control Room	
	Dose	*Criterion*	Dose	*Criterion*	Dose	*Criterion*
LOCA	12.2	*25*	2.99	*25*	4.00	*5*
MSLB Pre-accident Acccident-initiated	 0.127 0.175	 *25* *2.5*	 0.073 0.406	 *25* *2.5*	 0.790 4.255	 *5* *5*
CREA	4.647	*6.3*	1.983	*6.3*	2.549	*5*
LRA	1.456	*2.5*	0.525	*2.5*	2.529	*5*
SGTR Pre-accident Accident-initiated	 0.721 0.327	 *25* *2.5*	 0.165 0.077	 *25* *2.5*	 0.760 0.196	 *5* *5*
FHA 48-hr decay	 4.24	 *6.3*	 0.356	 *6.3*	 4.55	 *5*

Table 4

Analysis Assumptions for Loss of Coolant Accident

Parameter	**Value**
Power level, MWt	3,658.3
Source term model	RG 1.183
Primary containment free volume, ft^3	2,850,000
Sprayed volume of containment, percent	82.5
Maximum spray delay time, sec	17.5
Spray iodine removal coefficient, hr^{-1}	
Elemental	
Until DF=100 (t = 1.926 hr)	20
Organic	0
Particulate/Aerosol	
Until DF = 50 (t = 2.21 hr)	6.0
DF > 50 (2.21 - 8 hrs)	0.6
Mixing rate between sprayed and unsprayed regions, cfm	130,000
Primary containment leakage, volume percent per day	
0 to 24 hours	0.2
>24 hours	0.1
Iodine natural deposition, particulate/aerosol only	Powers 10th Percentile
ECCS leakage, cc/hr	276,000
ECCS leakage flashing fraction	0.10
ECCS iodine release species, percent	
Elemental	97
Organic	3
ECCS recirculation start, min	11.6
Auxiliary building filter efficiency, percent	
Elemental and organic iodine	90
Particulates	99
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

Table 5

Analysis Assumptions for Fuel Handling Accident

Parameter	Value
Power level, MWt	3,658.3
Radial peaking factor	1.7
Number of fuel assemblies in core	193
Number of fuel assemblies damaged	1
Reactor shutdown time before fuel movement, hr (without containment or mitigation)	48
Core fractions released from damaged rods	
I-131	0.08
Other halogens	0.05
Kr-85	0.10
Other noble gases	0.05
Alkali metals	0.12
High burnup fuel multiplier	2
Iodine effective pool decontamination factor	200
Iodine species in building, percent	
Elemental	57
Organic	43
Duration of release, hr	2
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

Table 6

Analysis Assumptions for Control Rod Ejection Accident

Parameter	Value
Power level, MWt	3,658.3
Core fuel rods failed, percent	10
Radial peaking factor	1.7
Fraction of core fission product inventory from failed fuel rods	
Iodines and noble gases	0.10
Alkali metals	0.24
Core fuel rods melted, percent	0.25
Fraction of core fission product inventory from melted fuel	
Iodine	0.5
Noble gases	1.0
Alkali metals	0.5
<u>Secondary Steaming Pathway</u>	
Primary-to-secondary leak, gpm	1.0
SG steaming partition factor	
Iodine	0.1
Alkali metals	0.0055
Iodine chemical form, percent	
Elemental	97
Organic	3
Duration of release, hr	1.111
Steam release through SGs, lbm	
0-200 sec	600,000
200-4000 sec	1,900,000
<u>Containment Leakage Pathway</u>	
Containment leak rate, volume percent per day	
0 to 24 hr	0.2
>24 hr	0.1
Iodine chemical form, percent	
Elemental	4.85
Organic	0.15
Particulate	95
Particulate iodine natural deposition	Powers 10th Percentile
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

Table 7

Analysis Assumptions for Locked Rotor Accident

Parameter	Value
Power level, MWt	3,658.3
Core fuel rods failed, percent	2
Radial peaking factor	1.7
Core fractions released from damaged rods	
I-131	0.08
Other halogens	0.05
Kr-85	0.10
Other noble gases	0.05
Alkali metals	0.12
High burnup fuel multiplier	2
Primary-to-secondary leak rate, gpm	
Unaffected SGs (total)	0.654
SG with failed PORV	0.5
SG steaming partition factor	
Iodine	0.1
Alkali metals	0.0055
Iodine chemical form, percent	
Elemental	97
Organic	3
Duration of release, hr	40
Steam release through SGs, total lbm	
0-2 hr	719,000
2-8 hr	1,109,000
8-40 hr	2,664,000
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

Table 8

Analysis Assumptions for MSLB Outside Containment

Parameter	Value
Power level, MWt	3,658.3
Accident induced iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131*	1
Accident induced iodine spiking factor	500
Pre-existing iodine spike	
Primary coolant iodine concentration, μCi/gm DEI-131	60
Primary-to-secondary leak rate, gpm	
Unaffected SGs (total)	0.654
Faulted SG	0.5
SG steaming iodine partition factor	0.1
Iodine chemical form, percent	
Elemental	97
Organic	3
Duration of release, hr	40
Steam release through faulted SG, lbm	167,000
Steam release through intact SG, lbm	
0-2 hr	442,000
2-8 hr	977,000
8-40 hr	2,216,000
Primary coolant volume, gm	2.063×10^{8}
Secondary coolant volume, gm	
Intact SGs (3)	1.017×10^{8}
Faulted SG	7.575×10^{7}
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

*Dose Equivalent I-131

Table 9

<u>Analysis Assumptions for Steam Generator Tube Rupture Accident</u>

Parameter	**Value**
Power level, MWt	3,658.3
Accident induced iodine spike	
Primary coolant iodine concentration, µCi/gm DEI-131*	1
Accident induced iodine spiking factor	335
Pre-existing iodine spike	
Primary coolant iodine concentration, µCi/gm DEI-131*	60
Primary-to-secondary leak rate, gpm	
Unaffected SGs (total)	0.654
Faulted SG	0.5
SG steaming iodine partition factor	0.1
Iodine chemical form, percent	
Elemental	97
Organic	3
Duration of release, hr	8
Initial coolant masses, lbm	
Ruptured SG	70,300
Intact SGs	211,000
RCS	502,000
Break release to ruptured SG	Listed in February 15, 2005 letter, Attachment 6, Table 8.1
Steam release through intact SGs	Listed in February 15, 2005 letter, Attachment 6, Table 8.1
Atmospheric dispersion values, offsite receptors	Table 2
CR parameters	Table 10
CR atmospheric dispersion values	Table 1

*Dose Equivalent I-131

Table 10

Control Room Analysis Assumptions

Parameter	Value
CR volume, ft^3	200,000
Recirculation flow rate, cfm	43,500 ± 10 percent
Recirculation filter efficiency, percent	
Elemental	90
Organic	90
Particulate	80
Emergency makeup flow rate, cfm	6,000 ± 10 percent
Makeup filter efficiency, percent	
Elemental	95
Organic	95
Particulate	99
Delay to switch to emergency mode, min	30
Unfiltered inleakage assumption, cfm	1,000
CR occupancy factors	
0-24 hr	1.0
1-4 days	0.6
4-30 days	0.4
CR breathing rate, m^3/sec	3.5×10^{-4}

(4) 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) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 147 And the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

(6) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's Fire Protection Report, and as approved in the SER dated February 1987 through Supplement No. 8, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(3) 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) 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) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 147, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Deleted.

(4) Deleted.

(5) Deleted.

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

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 140, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

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 are 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 and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3586.6 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 140, and the Environmental Protection Plan contained in Appendix B, both of which are attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.