March 18, 2004

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

SUBJECT: ISSUANCE OF AMENDMENTS RE: ONE-TIME CHANGE TO THE COMPLETION TIME FOR RESTORATION OF A UNIT SPECIFIC ESSENTIAL SERVICE WATER TRAIN (TAC NOS. MB9547, MB9548, MB9545, AND MB9546)

Dear Mr. Crane:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 136 to Facility Operating License No. NPF-37 and Amendment No. 136 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 130 to Facility Operating License No. NPF-72 and Amendment No. 130 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to Exelon Generation Company, LLC's (Exelon's) application dated June 11, 2003. Additional information was provided in Exelon's submittals of December 5, December 30, 2003, and February 18, 2004. The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

The amendments revise the technical specifications (TS) 3.7.8 to approve an one-time change in the completion time from 72 hours to 144 hours required to restore an unit specific essential service water train to operable status. The one-time change is applicable to Braidwood Station, Unit 1 and both units at Byron Station.

Exelon's application dated June 11, 2003 proposed adding a license condition to Appendix C, Additional Conditions, for each license. However, in discussions with members of the Exelon staff, it was concluded that because the change revises the completion time specified in TS 3.7.8, a change to Appendix A, Technical Specifications, was more appropriate. By letter dated

February 18, 2004, Exelon acknowledged that TS 3.7.8 would be revised. This change is administrative in nature and does not change any requirements for operation of the plants.

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/

George F. Dick, Jr. , Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 136 to NPF-37
2. Amendment No. 136 to NPF-66
3. Amendment No. 130 to NPF-72
4. Amendment No. 130 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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5. Safety Evaluation

cc w/encls: See next page

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ADAMS Accession Number: ML040610869 (Amendment)
ADAMS Accession Number: ML040620062 (Byron Amdmt 136 TS)
ADAMS Accession Number: ML040620068 (Braidwood Amdmt 130 TS)

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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. 136
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 June 11, 2003, as supplemented on December 5, December 30, 2003, and February 18, 2004, 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. 136 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JHopkins for/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 18, 2004

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 136
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 June 11, 2003, as supplemented on December 5, December 30, 2003, and February 18, 2004, 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. 136 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JHopkins for/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 18, 2004

ATTACHMENT TO LICENSE AMENDMENT NOS. 136 AND 136

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

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

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

Remove Pages	Insert Pages
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
3.7.8-3	3.7.8-3

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 June 11, 2003, as supplemented on December 5, December 30, 2003, and February 18, 2004, 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. 130 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JHopkins for/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 18, 2004

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 June 11, 2003, as supplemented on December 5, December 30, 2003, and February 18, 2004, 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. 130 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by JHopkins for/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 18, 2004

ATTACHMENT TO LICENSE AMENDMENT NOS. 130 AND 130

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

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

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

Remove Pages	Insert Pages
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
3.7.8-3	3.7.8-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 130 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 dated June 11, 2003 (Reference 9.1), as supplemented by letters dated December 5, (Reference 9.2), December 30, 2003 (Reference 9.3), and February 18, 2004 (Reference 9.4), the Exelon Generation Company, LLC (Exelon or the licensee) requested an amendment to Appendix C, "Additional Conditions," of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment would allow an increase in the Completion Time (CT) of Required Action A.1, "Restore unit-specific SX train to OPERABLE status," associated with Technical Specification (TS) Section 3.7.8, "Essential Service Water (SX) System," from 72 hours to 144 hours. This would be a one-time extension of the CT to facilitate replacement of the SX pump suction isolation valves at Byron and Braidwood. References 9.2, 9.3, and 9.4 provided clarifying information within the scope of the original application, and did not change the initial proposed no significant hazards consideration determination.

1.2 Description of Licensee's Request

The proposed change would increase, on a limited basis, the CT to restore an inoperable SX train. Specifically, the proposed changes would revise the CT for Required Action A.1, "Restore unit-specific SX train to OPERABLE status," associated with TS Section 3.7.8, from 72 hours to 144 hours. For Braidwood Station, the CT extension may be invoked only once for Unit 1, while Unit 2 is in Mode 5 or 6 or is defueled, and remains applicable through the completion of Unit 2 Refueling 11. For Byron Station, the CT extension may be invoked for Unit 1, while Unit 2 is in Mode 5 or 6 or is defueled, and remains applicable through the completion of Unit 2 Refueling 11, and for Unit 2, while Unit 1 is in Mode 5 or 6 or is defueled, and remains applicable through the completion of Unit 1 Refueling 13. Because the "A" Train SX pump suction valves have already been replaced, another application of the CT extension was not requested for Braidwood 2 during the Unit 1 outage.

The licensee indicated that it requested the changes in order to replace the SX pump suction valves used for pump isolation from the SX water supply. Currently, due to long-term wear, the suction isolation valves for the 1B and 2B SX pumps at Braidwood Station and all SX pump (i.e., the 1A, 2A, 1B, and 2B pumps) suction isolation valves at Byron Station are degrading such that individual pump isolation on a specific unit may not be adequate to perform pump maintenance or downstream system component maintenance. In order to replace these suction isolation valves, the common upstream suction isolation valve for the 1A and 2A SX pumps (or the common upstream suction isolation valve for the 1B and 2B SX pumps) must be closed and the suction header drained. After draining the common suction header, both the Unit 1 A(B) and Unit 2 A(B) suction isolation valves will be replaced. This evolution is time consuming and maintenance history has shown that completion of the needed SX suction isolation valve replacement cannot be assured within the existing 72-hour CT window.

Replacement of the SX suction isolation valve will be conducted during a refueling outage; however, due to the configuration of the SX system, closing the common suction isolation valve for the 1A and 1B (or 1B and 2B) SX pumps, puts the operating unit into a 72-hour Limiting Condition for Operation (LCO). Consequently, not being able to complete the suction isolation valve replacement in the 72-hour CT, would result in the operating unit also being shut down or not completing this work to improve the material condition of the plant.

In the letter of February 18, 2004, the licensee referenced a telephone call with the staff in which it was agreed that it was appropriate to revise TS 3.7.8 rather than issue a license condition. The licensee's letter acknowledged the change.

2.0 REGULATORY EVALUATION

2.1 Description of System/Component and Current Requirements

The SX system is discussed in the Updated Final Safety Analysis Report (UFSAR), Section 9.2.1.2, "Essential Service Water System."

The SX system provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation and a normal shutdown, the SX system also provides this function for various safety related and non-safety related components.

The unit-specific SX system consists of two separate, electrically independent, 100 percent capacity, safety related, cooling water trains. Each train consists of a 100 percent capacity pump, piping, valving, and instrumentation. The pumps and valves are remotely and manually aligned, except in the unlikely event of a Loss of Coolant Accident (LOCA). The pumps are automatically started upon receipt of a safety injection signal or an undervoltage on the Engineered Safety Features (ESF) bus, and all essential valves are aligned to their post accident positions. The SX system is also the backup water supply to the auxiliary feedwater system.

The SX system includes provisions to crosstie the trains (i.e., unit-specific crosstie), as well as provisions to crosstie the units (i.e., opposite-unit crosstie). The opposite-unit crosstie valves (i.e., 1SX005 and 2SX005) must both be open to accomplish the opposite-unit crosstie. The system is normally aligned with the unit-specific crosstie valves open and the opposite-unit crosstie valves closed.

Each full-capacity SX system loop in each unit is supplied by a single pump rated at 24,000 gpm at 180 feet ±10 percent total developed head. Actual system flow varies with system lineup and conditions. UFSAR Tables 9.2-1, "Essential Service Water Heat Loads," and Table 9.2-11, "Essential Service Water Component Nominal Design Flow Rates," list the components served and the nominal rated component flows. The pumps are located on the lowest level of the auxiliary building to ensure the availability of sufficient net positive suction head. Emergency power is available to each pump from its respective ESF bus as shown in UFSAR Table 8.3-5, "Loading on 4160-Volt ESF Buses," and described in UFSAR Section 8.3.1, "Onsite AC Power Systems." At Byron Station, the suction supply is by one supply line running from each of the two redundant essential service mechanical draft cooling towers to the auxiliary building. Each supply line supplies one SX pump in each unit; each of the two pumps in a given unit takes its suction from a separate supply line. At Braidwood Station, the suction supply is by two intake lines running from the Safety Category I portion of the lake screen house essential pond to the auxiliary building. Each intake line supplies one SX pump in each unit; each of the two pumps in a given unit takes its suction from a separate intake line. The system, therefore, meets the single-failure criterion as shown in the analysis in UFSAR Table 9.2-2, "Single-Failure Analysis of the Essential Service Water System," for Braidwood Station, and UFSAR Tables 9.2-2 and 9.2-16, "Single Failure Analysis of the Ultimate Heat Sink," for Byron Station. At Byron Station, heat rejection from the SX system is to the SX cooling towers, both on a normal and on an emergency basis. The discharges from each loop in each unit are separate and fed to two separate and redundant return lines for return to the towers. The two discharges from each unit and the two return lines to the towers are arranged similar to the intakes, i.e., the two discharges from each unit run into separate return lines, and each return line is fed from one discharge from each unit. The single failure criterion is met as shown in UFSAR Tables 9.2-2 and 9.2-16. At Braidwood Station, heat rejection from the SX system is to the essential cooling pond, both on a normal and on an emergency basis. The discharges from each loop in each unit are separate and fed to two separate and redundant return lines for return to the pond. The two discharges from each unit and the two return lines to the pond are arranged similar to the intakes, i.e., the two discharges from each unit run into separate return lines, and each return line is fed from one of the two discharges from each unit. The single-failure criterion is met as shown in UFSAR, Table 9.2-2. The essential cooling pond is more fully discussed in UFSAR, Section 9.2.5, "Ultimate Heat Sink." At Byron Station, the SX cooling towers are designed to accommodate the heat load from both units simultaneously under both normal and accident conditions. The SX cooling towers and their auxiliary systems are more fully discussed in UFSAR Section 9.2.5.

The design basis of the SX system is for one SX train, in conjunction with the Component Cooling (CC) system and a 100 percent capacity containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the UFSAR, Section 6.2, "Containment Systems." This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the reactor coolant system by the emergency core cooling system pumps. The SX system is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SX system, in conjunction with the CC system, also cools the unit from Residual Heat Removal (RHR) entry conditions, as discussed in the UFSAR, Section 5.4.7, (i.e., Reference 2) to Mode 5 during normal and post accident operations. The time needed for this evolution is a function of the number of CC and RHR System trains that are operating. One SX train is sufficient to remove decay heat during subsequent operations in Modes 5 and 6.

2.2 Applicable Regulatory Requirements and Review Criteria

The NRC staff finds that the licensee has for the most part identified in Attachment 1, Section 5, of its submittal (Reference 5.1) the regulatory requirements that are applicable to this License Amendment Request. NRC requirements and review criteria that the staff considered to be most applicable include:

- 10 CFR 50.36, “Technical Specifications,” paragraph (c)(2)(ii)(C), Criterion 3, which requires that TS LCO be established for: “A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate the design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.”

- Standard Review Plan (SRP), Chapter 16.1, "Risk-Informed Decisionmaking: Technical Specifications," contains five key principles of the staff’s philosophy of risk-informed decisionmaking. They are: (1) the proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change; (2) The proposed change is consistent with the defense-in-depth philosophy; (3) the proposed change maintains sufficient safety margins; (4) when proposed changes result in an increase in core-damage frequency or risk, the increases should be small and consistent with the intent of the Commission’s Safety Goal Policy Statement; and (5) the impact of the proposed change should be monitored using performance measurement strategies. The staff recognizes that the proposed change to the SX CT is a temporary change and that the criteria are for permanent changes. Therefore the staff used the criteria to guide its review and applied them to the extent possible.

- Regulatory Guide 1.174 (RG 1.174), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

- Regulatory Guide 1.177 (RG 1.177), "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," describes an acceptable risk-informed approach specifically for assessing proposed TS changes in allowed outage times (AOTs). Note that the phrase “completion time” used in the licensee’s TS is equivalent to the phrase “allowed outage time” used in RG 1.177. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

It should be noted that RG 1.174 and RG 1.177 are directly applicable only to permanent (as opposed to temporary, or “one-time”) changes to TS requirements. However, the staff has previously consulted these regulatory guides while making risk-informed decisions about temporary TS changes. In the context of risk-informed decisionmaking about TS changes, the risk acceptance guidelines in RG 1.174 and RG 1.177 are not applied in an overly prescriptive manner; rather, they provide an indication, in numerical terms, of what is considered acceptable. The intent in comparing risk results with the risk acceptance guidelines is to demonstrate with reasonable assurance that increases in risk caused by proposed changes are small and consistent with the Commission’s Safety Goal Policy.

The staff’s review of the licensee’s proposed change includes operator actions, human-system interfaces, procedures, and training related to the proposed change. The NRC’s acceptance criteria for operator performance are based on GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

3.0 TECHNICAL EVALUATION

3.1 Defense-In-Depth Evaluation

The elements of the defense-in-depth philosophy are referred to by RG 1.177 and described in RG 1.174. Reference 5.1, Attachment 1, Section 4.1, Table 1, provides the licensee’s defense-in-depth assessment of the planned SX configuration; and Table 2 provides an evaluation of the RG 1.174 defense-in-depth considerations for the proposed one-time extension of the TS 3.7.8 CT for Required Action A.1. The more substantive defense-in-depth considerations include:

- Based on the “best estimate analysis” (from Table 1 referred to above), the licensee has determined that one SX pump from either unit is capable of supporting the shutdown loads of both units, with the exception of the reactor containment fan coolers (RCFC’s) and emergency diesel generators (EDGs). Therefore, redundancy is maintained for most shutdown situations that do not involve a loss-of-offsite power.

- Defense against potential common cause failures are preserved, since no new common cause failure mechanisms are introduced by this change.

- While the licensee has identified compensatory measures to assure that the defense-in-depth capability is maintained during the period when the extended CT is in effect, these measures are consistent with normal plant practice and do not constitute an over-reliance on programmatic activities.

- Defense against human errors is preserved to some extent by establishing a dedicated operator to monitor SX status (see compensatory measures, below).

Additional defense-in-depth capabilities applicable to the Byron and Braidwood stations include provisions to allow: (1) alignment of the fire protection system to provide makeup water for the chemical and volume control (CV) centrifugal charging pumps for reactor coolant pump seal cooling, and (2) operation of the diesel-driven auxiliary feedwater (AFW) pumps and the startup and motor-driven main feedwater pumps without SX flow, which allows the plant to maintain hot standby conditions if a complete loss of all SX cooling should occur on the operating unit.

In order to assure that the defense-in-depth capabilities referred to above are maintained during the period when the extended CT is in effect, the licensee has committed to implement the following compensatory measures:

+ Restrict maintenance on the component cooling heat exchangers including the opposite-unit crosstie valves
+ Restrict maintenance on the AFW system
+ Restrict maintenance on the station auxiliary transformers
+ Restrict maintenance on appropriate electric power equipment, including key 4KV buses, EDGs, and battery chargers
+ Isolate appropriate RCFCs
+ Provide a “dedicated” operator to monitor SX status and respond to any SX-related problems (e.g., cross-connect SX from the shutdown unit)

These will be discussed further in Section 4.0, Regulatory Commitments of this Safety Evaluation.

3.1 Summary

Based on a review of the information that was provided, as summarized above, the staff finds that the defense-in-depth considerations and compensatory measures committed to by the licensee and discussed above satisfy the RG 1.174 criteria for making risk-informed changes. In particular, even though the proposed License Condition will result in reduced redundancy for an additional 72 hour period beyond what is normally allowed by TS 3.7.8, the operable SX pump of the shut down unit provides additional limited capability that is in excess of what is normally needed for a 72 hour CT; and the licensee has demonstrated that the Byron and Braidwood units possess additional cooling capability beyond that considered and relied upon in the design basis. Because the proposed change will only apply on a one-time basis, will not result in a permanent change to the TS requirements, the staff considers the defense-in-depth capability credited by the licensee and discussed above to be appropriate and acceptable.

3.2 Operator Performance

The licensee provided information regarding credit for additional operators as a key compensatory measure for the proposed amendment in its June 11, 2003 application and in its December 5, 2003 responses to the staff’s request for additional information. The following is a summary of the licensee’s responses and the NRC staff’s conclusions regarding the area of operator performance:

3.2.1 Staffing

With regard to increasing staffing, the licensee indicated that “dedicated” and “designated” operators will be utilized in addition to the minimum staffing levels required by the TS and by 10 CFR 50.54, “Conditions of licenses,” paragraph (m)(2)(i). The staff position on “dedicated” is for the operator to be located in the immediate vicinity of where the task needs to be performed and to be capable of performing the task on demand, requiring no decision time, preparation time, or travel time. "Dedicated" does not necessarily mean that the operator cannot have other duties, as long as the other duties do not interfere with performing the required tasks. The staff position on "designated" is for the operator to be available to perform the specific task when required, but he/she may have other responsibilities away from the immediate vicinity of where the required task is to be performed. Additionally, both "dedicated" and "designated" imply that the individual should be "qualified" to perform the task.

In response to the above descriptions, the licensee indicated that a senior reactor operator (SRO) and a reactor operator (RO) will be considered to be the "dedicated" operators, and an equipment operator (i.e., non-licensed operator) will be the "designated" operator, with assigned tasks that can be deferred immediately if designated actions are necessary. These personnel will be assigned both inside and outside the control room to monitor SX performance and to provide back up to the nominal staff, and all will use the standard three way communication protocol.

In addition, the dedicated SRO and RO will monitor SX performance and report directly to the operating Unit Supervisor, who would then report to the Shift Manager. The dedicated SRO and RO will inform the Unit Supervisor if they identify that entry conditions have been reached and the associated abnormal operating procedures have been implemented and provide reports on the results of the actions. The designated equipment operator will report to the dedicated SRO. Because the dedicated and designated operators' duties conform to the criteria of the staff positions above, the staff finds the licensee's additional staffing commitment and the command, control, and communication arrangement with the additional operators to be satisfactory.

3.2.2 Operator Actions

The licensee provided a list of three tasks anticipated to be performed by the additional personnel. These tasks included establishing SX unit crosstie, establishing alternate cooling to CV pump oil cooler, and establishing cool suction source for CV pumps. Additional information such as location of tasks and task complexity/difficulty were provided in sufficient detail for the staff to judge that the tasks can be successfully accomplished.

The licensee also provided the time necessary to perform each of the three tasks, with travel time included. The licensee's estimates were based on simulator training records and job performance measures, with travel times estimated based on engineering judgment. All times necessary to accomplish the tasks were well within the time available. The staff finds the time available for operators to successfully perform each task to be sufficient and is satisfied that the tasks are well within the operators' capability.

3.2.3 Training and Procedures

The licensee indicated that control room and equipment operators will receive refresher training on procedures related to establishing SX unit crosstie and alternative cooling for the CV pumps prior to entering the CT. Briefings will also be provided to ensure familiarity with the appropriate actions, location of equipment, tools necessary to execute tasks, and to ensure understanding of the entry conditions for the compensatory actions.

Prior to each shift change, the licensee will provide copies of procedures for the action to align fire protection cooling to the designated equipment operator. This should minimize the need to locate the procedure during conditions of SX loss. The staff finds the licensee's commitments to training and providing procedures to be acceptable.

3.2.4 Equipment and Environmental Conditions

The licensee indicated that ladders/stepping stools will be placed in locations where needed for the designated equipment operator to reach the fire protection hose hookup without assistance. This should minimize the time needed for the operator to carry out the specified action.

Additionally, the licensee stated that equipment to be used by the control room to communicate with designated equipment operator will be normal radio systems, telephone, or plant page. These systems are normally used frequently, and therefore, based on operational experience, they can be verified to be operational and proven to be adequate.

The licensee stated that with the exception of the Fire Protection connections to the CV pumps, all actions can be taken from the control room. The CV pump rooms, located in the auxiliary building, are not in areas in which environmental conditions preclude access and will not have special access requirements other than normal radiation work permit compliance. The staff finds the equipment use and environmental conditions described by the licensee to be acceptable.

3.2.5 Summary

The NRC staff has reviewed the licensee's planned use of additional operators as a compensatory measure. The staff concludes that the licensee has adequately considered the impact of the proposed license amendment on operator staffing, procedures, equipment, and associated training to provide reasonable assurance for allowing credit for the proposed actions to be performed by the additional operators. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR Part 55, and will continue to follow the guidance of GL 82-33. Therefore, the NRC staff finds the licensee's proposed License Amendment acceptable with respect to the operator performance aspects of the proposed change.

3.3 Risk Evaluation

3.3.1 Key information Used in Staff Review

The key information used in the staff's review of the risk evaluation is contained in Attachment 1, Section 4.3 and Attachment 4 to the licensee's submittal (Reference 9.1), as supplemented by the licensee in response to the staff's Request for Additional Information (Reference 9.2). In additional, the staff consulted the Safety Evaluation Reports (SERs) on the Individual Plant Examinations (IPEs) and Individual Plant Examinations - External Events (IPEEEs) submitted by the licensee (References 9.8 through 9.11).

The risk evaluation presented below addresses the last two key principles of the staff's philosophy of risk-informed decision making, which concern changes in risk and performance measurement strategies. These key principles were evaluated by using the three-tiered approach described in Chapter 16.1 of the SRP and RG 1.177.

- Tier 1 - The first tier evaluates the licensee's probabilistic risk/safety assessment (PRA/PSA) and the impact of the change on plant operational risk, as expressed by the change in core damage frequency (CDF) and the change in large early release frequency (LERF). The change in risk is compared against the acceptance guidelines presented in RG 1.174. The first tier also aims to ensure that plant risk does not increase unacceptably during the period when equipment is taken out of service per the license amendment, as expressed by the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). The incremental risk is compared against the acceptance guidelines presented in RG 1.177.

- Tier 2 - The second tier addresses the need to preclude potentially high-risk plant configurations that could result if equipment, in addition to that associated with the proposed license amendment, are taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The objective of this part of the review is to ensure that appropriate restrictions on dominant risk-significant plant configurations associated with the AOT extension are in place.

- Tier 3 - The third tier addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and taking appropriate compensatory measures to avoid such configurations. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended AOT period will be appropriately assessed from a risk perspective.

3.3.2 Tier 1: PRA Capability and Insights

The Tier 1 staff review involved two aspects: (1) evaluation of the validity of the PRA and its application to the proposed CT extension and (2) evaluation of the PRA results and insights stemming from its application.

3.3.2.1 Evaluation of PRA Validity

To determine whether the PRA used in support of the proposed CT extension is of sufficient quality, scope, and level of detail, the staff evaluated the relevant information provided by the licensee in their submittal, as supplemented, and considered the findings of recent PRA reviews. The staff's review of the licensee's submittal focused on the validity of the licensee's PRA model to analyze the risks stemming from the proposed CT extension and did not involve an in-depth review of the licensee's PRA.

Braidwood Station and Byron Station share a single PRA model that contains appropriate logic flags (house events) and databases to differentiate between units and sites. The PRA model addresses internal events at full power, including anticipated transients without scram (ATWS), interfacing system LOCAs outside the containment (ISLOCAs), and internal floods. It is an evolution of the original IPEs submitted in response to Generic Letter 88-20 as follows:

Station	Date of IPE Submittal	Date When SER Issued
Braidwood	June 30, 1994	October 27, 1997
Byron	April 28, 1994	December 3, 1997

The most recent update of the PRA model (completed May 16, 2003) reflects the as-built, as-operated plant configuration as of December 31, 2002. In addition to incorporating recent advances in PRA technology and various plant modifications, special effort was made to ensure that elements of the PRA model that are sensitive to changes in SX pump maintenance unavailability were adequate to evaluate the risk impacts of the increased CT for the SX pump trains. These elements include:

- The impact of maintenance unavailability on the loss of SX initiating event frequencies
- Treatment of the SX unit and train crosstie capability
- Use of plant-specific data
- Operator interviews to validate appropriate characterization of operator actions available to mitigate a loss-of-SX initiating event

The PRA model uses a plant-specific model that is an extension of the methodology described in NUREG/CF-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," January 1999, to estimate the LERF. The LERF model is based on:

- Plant damage states that reflect the appropriate plant conditions at the time of core damage
- Containment isolation failures based on plant-specific system models
- Plant-specific analysis of containment bypass sequences such as ISLOCAs and steam generator tube ruptures (SGTR)
- Realistic, plant-specific assessment of unit-specific containment ultimate pressure capability

- Important severe accident phenomena including high-pressure melt ejection and induced SGTR
- Plant-specific thermal-hydraulic analysis reflecting the severe accident conditions expected to be present

Braidwood and Byron Stations have evaluated the risk of external hazards as part of the IPEEE process. The dates when each IPEEE were submitted and the associated SER issued are shown below:

Station	Date of IPEEE Submital	Date When SER Issued
Braidwood	June 27, 1997	May 30, 2001
Byron	December 23, 1996	May 30, 2001

The analysis of risks due to internal fires was based on the Fire-Induced Vulnerability Evaluation method developed by the Electric Power Research Institute (EPRI). The seismic margins analysis (SMA) method was used to assess risks due to seismic events. A screening analysis of other external events (such as transportation accidents) revealed no potential vulnerabilities. Since the IPEEEs were completed, a number of plant improvements (such as the addition of capability to cool the centrifugal charging pumps using water for the fire protection system) have been implemented. However, the IPEEE analyses have not been updated to reflect the as-built, as-operated plants. The risks from external events included with the licensee's submittal are based on qualitative assessments using the insights of the internal events PRA. The staff finds this approach acceptable since the licensee has demonstrated that the proposed one-time extension of the SX train CT does not unacceptably degrade plant capability during external events, and does not introduce risk vulnerabilities or remove elements of the plant response capability from programmatic activities aimed at ensuring satisfactory safety performance during external hazards.

There is no shutdown PRA model for the Braidwood and Byron Stations.

In order to help ensure the validity of the PRA, an independent assessment was completed by an outside industry expert in July, 1999, using the self-assessment process developed as part of the Westinghouse Owners Group (WOG) PRA Peer Review Certification Program. Peer review certification of the Braidwood Station PRA using the WOG Peer Review Certification Guidelines was completed in August 1999. Certification of the Byron Station PRA was performed in July 2000. PRA model revision since the peer reviews have been independently reviewed by the Braidwood and Byron Site Risk Management staff. The licensee has reviewed all open items arising from the reviews identified above. Each open item was dispositioned by making changes during the recent PRA model update or determining that the open item did not affect the conclusions of the PRA.

In 2000, the staff assessed the quality of the PRA used in support of a license amendment to extend the CT of the EDGs at Braidwood and Byron Stations. The SER attached to the issuance of the license amendment, issued September 1, 2000, indicates that the risk analysis used in support of the license amendment was of sufficient quality.

Based on the above information, the staff finds that the licensee has satisfied RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), and SRP Chapter 19.1, and that the quality of the Braidwood/Byron PRA is sufficient to support the risk evaluation provided by the licensee in the proposed license amendment.

3.3.2.2 Evaluation of PRA Results and Insights

The licensee has calculated the following risk metrics, which include the contributions from internal initiating events and internal floods:

Risk Metric	Braidwood Unit 1	Byron Unit 1	Byron Unit 2
baseline core-damage frequency (CDF)	6E-5/y	6E-5/y	6E-5/y
CDF based on 144-hour SX train CT including compensatory measures	8E-5/y	8E-5/y	8E-5/y
ICCDP including compensatory measures	3E-7	3E-7	3E-7
baseline LERF	5E-6/y	5E-6/y	6E-6/y
LERF based on 144-hour SX train CT including compensatory measures	5E-6/y	5E-6/y	7E-6/y
ICLERP including compensatory measures	7E-9	5E-9	1E-8

The licensee has indicated that the following compensatory measures were incorporated into the risk evaluation of internal events:

- Restrict maintenance on component cooling (CC) system heat exchangers, including the opposite-unit SX crosstie valves.
- Restrict maintenance on auxiliary feedwater (AFW).
- Restrict maintenance on the station auxiliary transformers (SATs)
- Restrict maintenance on appropriate electrical power equipment, including key 4KV buses, emergency diesel generators (EDGs), and battery chargers
- Isolate appropriate reactor containment fan coolers (RCFCs)
- Provide dedicated operators to monitor SX and respond to any SX-related problems (e.g., cross-connect SX as needed).

The licensee did not revise the quantitative analysis of internal fire vulnerabilities that was conducted as part of the IPEEE process, which conservatively estimated the fire-induced core-damage frequencies to be:

Risk Metric	Braidwood Unit 1	Byron Unit 1	Byron Unit 2
baseline CDF due to internal fires, based on IPEEE process	4E-6/y	5E-6/y	6.1E-6/y

The licensee performed a qualitative assessment to identify the areas of the plant where the plant configuration during replacement of the SX pump suction isolation valves could lead to increased fire risks. The following process was used:

- Plant equipment and cable location data were reviewed to identify fire scenarios where the plant configuration during replacement of the SX pump suction isolation valves could result in an increase in fire risk.
- For each of these areas, the internal events PRA was used to identify whether a success path for achieving safe shutdown could be identified for that area.
- Compensatory measures, including those considered in the internal events analysis, were identified to reduce the likelihood of a fire in those areas and increase mitigation capability.

Based on the above information, the staff has concluded that the contribution of internal fires to the overall baseline CDF is small, and that the internal fire risk contribution is not appreciably impacted by the plant configuration during replacement of the SX pump suction isolation valves. Based on this conclusion, it is reasonable to infer that the contribution of internal fires to the overall baseline LERF is small. Further, the contribution of internal fires to the change in CDF (ΔCDF) and change in LERF (ΔLERF) caused by the plant configuration during replacement of the SX pump suction isolation valves is very small.

The licensee reviewed the SMA that was conducted as part of the IPEEE process, which determined that the plant-level high confidence of low probability of failure (HCLPF) is 0.3g. The interaction between seismic events and the SX system are dominated by seismically induced loss of offsite power (LOOP) and station blackout (SBO) events, as well as the function of the SX system to provide a suction source to the auxiliary feedwater pumps following a seismic event. The IPEEE process did not identify any seismic vulnerabilities; however, some plant improvements are being addressed:

- Control room ceiling diffusers are made of aluminum and, if dislodged by a seismic event, may pose a personnel hazard. A design change is being implemented to ensure that the diffusers are adequately restrained.

- There is a potential for seismic interactions between closely spaced electrical cabinets that contain essential relays. Plant walkdowns were completed and additional analysis is being performed to determine what, if any, corrective measures are necessary.

Plant modifications completed after the IPEEE were assessed by the licensee to determine if the assumptions behind the SMA were still valid. The licensee determined that the configuration of the SX system does not affect its seismic capacity. At Byron Station, a new security tower was erected and attached to the SX towers. This tower was designed to all design basis seismic, high wind, and tornado loadings. However, the design basis earthquake at the Byron site is 0.2g as compared to the IPEEE seismic event of 0.3g. The licensee assessed the seismic loading on the security tower from a 0.3g earthquake and determined it to be bounded by the tornado loading for the structure. Based on the above information, the licensee determined that the plant-level HCLPF has been maintained at 0.3g with the exceptions of seismically induced LOOP and SBO events.

The licensee has determined that the seismically induced dual-unit loss of offsite power (DLOOP) is about 4E-5/y for Byron Station and 3E-5/y for Braidwood Station, based on the seismic hazard curves from NUREG-1488 (Reference 7) and generic offsite power fragilities from the NUREG-1150 study (Reference 8). In comparison, the non-recoverable internal events DLOOP frequency is on the order of 1E-3/y. The licensee has determined that there are no issues regarding the plant configuration during replacement of the SX pump suction isolation valves that negatively impact the seismically induced DLOOP frequency. Therefore, during replacement of the SX pump suction isolation valves, the risk from seismically induced DLOOPs is very small compared to the risk from DLOOPs due to internal events, and is subsumed within the licensee's risk evaluation of the proposed one-time extension of the SX train CT.

The licensee has determined that the seismically induced SBO is about 1E-6/y for both Byron and Braidwood Stations, based on the seismic hazard curves from NUREG-1488 (Reference 7) and generic emergency diesel generator (EDG) fragilities from the NUREG-1150 study (Reference 8). Seismically induced LOOP with random SX pump failure would result in an SBO on the unit with the failed pump. In this case, the SX crosstie is not credited in the PRA since it is assumed that the EDG fails due to lack of cooling before the SX crosstie is made. Furthermore, the 4kV crosstie is not credited in the PRA when only one EDG is available on both units. Based on the internal events PRA, the probability of SX pump failure to start and run after a demand is about 2E-3. Given a seismically induced LOOP frequency of about 4E-5/y, the frequency of seismically induced LOOP with random SX pump failure is less than 1E-7/y, about an order of magnitude below the seismically induced SBO frequency. Therefore, during replacement of the SX pump suction isolation valves, the risk from seismically induced LOOP with random failure of the operating SX is very small compared to the risk from seismically induced SBO, and is subsumed within the licensee's IPEEE assessment of seismic risks. Based on the above information, the staff has concluded that the contribution of seismic events to the overall baseline CDF is small, and that the seismic event risk contribution is not appreciably impacted by the plant configuration during replacement of the SX pump suction isolation valves. Based on these conclusions, it is reasonable to infer that the contribution of seismic events to the overall baseline LERF is small. Further, the contribution of seismic events to the change in CDF (ΔCDF) and change in LERF (ΔLERF) caused by the plant configuration during replacement of the SX pump suction isolation valves is very small.

Based on the above assessment of fire and seismic risks during the proposed one-time extension of the SX train CT, the risk metrics that consider only contributions from internal initiators and internal flooding are reasonable approximations to the risk metrics that consider all contributors (including external events). Therefore, a comparison of the risk metrics that consider only contributions for internal initiators and internal flooding to the risk acceptance guidelines in RG 1.177 and RG 1.174 provides a sufficient basis for deciding whether or not the risk increases associated with the proposed one-time extension of the SX train CT are small and consistent with the intent of the Commission's Safety Policy Statement.

Section 2.4 of RG 1.177 states that a permanent TS CT change has only a small quantitative impact on plant risk if the incremental conditional core damage probably (ICCDP) is less than 5.0E-7 and the ICLERP is less than 5.0E-8. Comparison of the ICCDP and ICLERP values generated by the licensee (shown above) to the RG 1.177 risk acceptance guidelines indicates that the proposed 144-hour SX train CT has only a small plant risk impact.

Section 2.4 of RG 1.177 calls for the comparison of risk metrics to the risk acceptance guidelines contained in Section 2.2.4 (ΔCDF versus baseline CDF) and Section 2.2.5 (ΔLERF versus baseline LERF) of RG 1.174. The "delta risk" metrics (ΔCDF and ΔLERF) of RG 1.174 reflect the change in risk due to the increase in annualized SX pump unavailability associated with the one-time SX pump suction isolation valve replacement (6 days out of 365 days). Using the risk metrics provided by the licensee, the staff has estimated that the change in CDF (ΔCDF) is between 1E-6/y and 1E-5/y for all units. The estimated change in LERF (ΔLERF) is between 1E-7/y and 1E-6/y for all units. Considering the baseline CDF and LERF values previously stated, the staff concludes that the risk impact of the proposed one-time extension of the SX train CT lies in Region II of Figures 3 and 4 contained in RG 1.174.

Therefore, in accordance with the RG 1.174 risk acceptance guidelines, the licensee's proposed license amendment to allow a one-time extension of the SX train CT of 144 hours days results in an acceptable increase in risk which is small and consistent with the NRC's Safety Goal Policy Statement.

Therefore, the staff finds that the licensee's first tier risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.3.3 Tier 2: Avoidance of Risk-Significant Plant Configurations

The second tier evaluates the capability of the licensee to recognize and avoid risk-significant plant configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out of service simultaneously or if other risk- significant operational factors, such as concurrent system or equipment testing, are also involved.

The licensee stated that the evaluation of risk-significant plant configurations considers the impact both on the at-power unit for which the license amendment is requested and on the opposite unit which will be shut down during the planned configuration. The dual-unit aspects are addressed as part of the licensee's overall configuration risk management program (CRMP), which is discussed below in the third tier evaluation.

As discussed in the first tier evaluation, the licensee incorporated certain compensatory measures into the risk evaluation of internal events. Without these compensatory measures, the licensee estimated the ICCDP and ICLERP values to be:

Risk Metric	Braidwood Unit 1	Byron Unit 1	Byron Unit 2
ICCDP without any compensatory measures	6E-6	6E-6	6E-6
ICLERP without any compensatory measures	2E-7	2E-7	3E-7

Since these ICCDP and ICLERP values are well above the risk acceptance guidelines in RG 1.177, the licensee has made various regulatory commitments (Section 4.0) to implement the compensatory measures. The use of PRA methods to identify and evaluate the efficacy of compensatory measures demonstrates the licensee's ability to recognize and avoid risk-significant plant configurations. Therefore, the staff finds that the licensee's second tier risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.3.4 Tier 3: Risk-Informed Configuration Risk Management

Byron and Braidwood Stations have developed a CRMP governed by station procedures that ensures the risk impact of equipment out of service is appropriately evaluated prior to performing any maintenance activity. This program provides for an integrated review to uncover risk-significant plant equipment outage configurations in a timely manner both during the work management process and for emergent conditions during normal plant operation. Appropriate consideration is given to equipment unavailability, operational activities like testing or load dispatching, and weather conditions. Byron and Braidwood Stations currently have the capability to perform a configuration dependent assessment of the overall impact on risk of proposed plant configurations prior to, and during, the performance of maintenance activities that remove equipment from service. Risk is reassessed if an equipment failure/malfunction or emergent condition produces a plant configuration that has not been previously assessed.

For planned maintenance activities, an assessment of the overall risk of the activity on plant safety, including benefits to system reliability and performance, is currently performed prior to scheduled work. The assessment includes the following considerations:

- Maintenance activities that affect redundant and diverse structures, systems, and components (SSCs) that provide backup for the same function are minimized.

- The potential for planned activities to cause a plant transient are reviewed and work on SSCs that would be required to mitigate the transient are avoided.

- Work is not scheduled that is highly likely to exceed a TS or Technical Requirements Manual (TRM) Completion Time requiring a plant shutdown. For activities that are expected to exceed 50 percent of a TS Completion Time, compensatory measures and contingency plans are considered to minimize SSC unavailability and maximize SSC reliability.

- For Maintenance Rule high risk significant SSCs, the impact of the planned activity on the unavailability performance criteria is evaluated.

- As a final check, a quantitative risk assessment is performed to ensure that the activity does not pose any unacceptable risk. This evaluation is performed using the impact on both CDF and LERF. The results of the risk assessment are classified by a color code based on the increased risk of the activity.

The overall risk assessment is a part of the licensee's implementation of the Maintenance Rule, which provides the appropriate regulatory control.

Emergent work is reviewed by shift operations to ensure that the work does not invalidate the assumptions made during the work management process. The licensee's PRA risk management procedure has been implemented at Byron and Braidwood Stations. This procedure defines the methods for ensuring that the PRA model used to evaluate on-line maintenance activities is an accurate model of the current plant design and operational characteristics. Plant modifications and procedure changes are monitored, assessed, and dispositioned. Evaluation of changes in plant configuration or PRA model features are dispositioned by implementing PRA model changes or by the qualitative assessment of the impact of the change on the PRA assessment tool. Changes that have potential risk impact are recorded in an update log for consideration in the next periodic PRA model update.

Based on the above, the staff finds that the licensee's first third risk evaluation, as described in Chapter 16.1 of the SRP and RG 1.177, is acceptable.

3.3.5 Summary

In summary, the staff finds that the licensee's proposed changes to TS Section 3.7.8 to extend the SX train CT from 72 hours to 144 hours are acceptable because the acceptance guidelines of RG 1.174 and RG 1.177 are met.

4.0 REGULATORY COMMITMENTS

As noted previously, the licensee proposed a number of compensatory measures that will be taken to assure that risk impacts are acceptably low. The compensatory measures are:

- Restrict maintenance on component cooling (CC) system heat exchangers, including the opposite-unit SX crosstie valves.
- Restrict maintenance on auxiliary feedwater (AFW).
- Restrict maintenance on the station auxiliary transformers (SATs)
- Restrict maintenance on appropriate electrical power equipment, including key 4KV buses, emergency diesel generators (EDGs), and battery chargers
- Isolate appropriate RCFCs
- Provide dedicated operators to monitor SX and respond to any SX-related problems (e.g., cross-connect SX as needed).

In their letter of February 18, 2004, the licensee informed the staff that the compensatory measures are considered to be regulatory commitments.

The licensee is required under the Maintenance Rule (10 CFR 50.65(a)(4)) to assess and manage risk that may result from maintenance activities. As noted in Section 3.3.4 of this SE, the licensee has developed a CRMP which is used to assess the risk impact of equipment out of service as part of the maintenance work management process and for an evaluation of emergent conditions during normal plant operation. The regulatory commitments listed above are actions that will be taken by the licensee to reduce the risk associated with replacement of the SX pump suction valves, which are maintenance activities, and are covered by the Maintenance Rule. Therefore the licensee will evaluate the risk impact of any proposed changes to the commitments in accordance with their CRMP, which is one aspect of their program for implementing the Maintenance Rule. The staff considers this to be adequate control of the regulatory commitments.

5.0 TECHNICAL SPECIFICATIONS

In consideration of the nature of the licensee's request, the staff concluded that the result of approval of the licensee's request would be a limited duration extension of the CT specified in Condition A of TS 3.7.8 from 72 hours to 144 hours. Consequently, a revision to the TS would be more appropriate than adding a license condition. This approach was discussed with the licensee during a telephone conversation on February 5, 2004. There was agreement between the staff and the licensee that the staff would issue a revision to TS 3.7.8 rather than a revision to Appendix C, "Additional Conditions," of the licenses. This change from a license condition to a TS revision is administrative in nature and does not change the requirement on the licensee. However, the change does maintain the integrity of the format for the TS.

As a result, a note was added to Condition A of TS 3.7.8 stating that the 72-hour CT does not apply while one unit is in Mode 5 or 6, or is defueled, until the end of the stated refueling outage. Condition B was added to specify the limited use of the 144-hour CT for the SX suction valve replacement under those conditions. The former Condition B was changed to Condition C and the former Condition C was renamed as Condition D.

6.0 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.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 56342). 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.

8.0 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.

9.0 REFERENCES

9.1 Letter from Keith R. Jury (Director - Licensing, Midwest Regional Operating Group), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Request for a License Amendment for a One-Time Extension of the Essential Service Water Train Completion Time," dated June 11, 2003. (ML031700356)

9.2 Letter from Kenneth A. Ainger (Manager, Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding a License Amendment for a One-Time Extension of the Essential Service Water Train Completion Time," dated December 5, 2003.

9.3 Letter from Kenneth A. Ainger (Manager, Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Clarification of Compensatory Actions Associated with a License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated December 30, 2003.

9.4 Letter from Kenneth A. Ainger, Exelon Generation Company, LLC, to U.S. Nuclear Regulatory Commission, "Additional Information Regarding a License Amendment Request for a One-?Time Extension of the Essential Service Water Train Completion Time," dated February 18, 2004. (ML040540648)

9.5 Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998.

9.6 Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998.

9.7 Updated Final Safety Analysis Report for Byron and Braidwood Stations.

9.8 Letter from U.S. Nuclear Regulatory Commission to Irene Johnson, Commonwealth Edison Company, "Individual Plant Examination, Braidwood Units 1 and 2," October 27, 1997.

9.9 Letter from U.S. Nuclear Regulatory Commission to Irene M. Johnson, Commonwealth Edison Company, "Individual Plant Examination, Byron Station, Units 1 and 2," December 3, 1997.

9.10 Letter from U.S. Nuclear Regulatory Commission to Oliver D. Kingsley, Exelon Generation Company, “Review of Braidwood Individual Plant Examination of External Events,” May 30, 2001.

9.11 Letter from U.S. Nuclear Regulatory Commission to Oliver D. Kingsley, Exelon Generation Company, “Review of Individual Plant Examination of External Events - Byron Station, Units 1 and 2,” May 30, 2001.

9.12 NUREG-1488, “Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains,” April 1994.

9.13 NUREG/CR-4550, Vol. 3, Rev. 1, Part 3, “Analysis of Core Damage Frequency: Surry Power Station, Unit 1 External Events,” December 1990.

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