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

April 9, 2010

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BYRON STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: EXTENSION OF ESSENTIAL SERVICE WATER TRAIN TECHNICAL SPECIFICATION COMPLETION TIME (TAC NOS ME2293 AND ME2294)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 168 to Facility Operating License No. NPF-37 and Amendment No. 168 to Facility Operating License No. NPF-66 for the Byron Station (Byron), Unit Nos. 1 and 2, respectively. The amendments are in response to the Exelon Generation Company, LLC (EGC, the licensee) application dated September 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092680090), as supplemented by letters dated
November 13, 2009 (ADAMS Accession No. ML093200065), January 19, 2010 (ADAMS Accession No. ML100200075), March 1, 2010 (ADAMS Accession No. ML100610109), March 9, 2010 (two letters) (ADAMS Accession Nos. ML100700557 and ML100700613), and March 19, 2010 (ADAMS Accession Nos. ML100780401).

The amendments allow an extension of the Completion Time (CT) to restore a unit-specific essential service water (SX) train to operable status associated with the Limiting Condition for Operation for Technical Specification 3.7.8, "Essential Service Water (SX) System," from 72 hours to 144 hours, only during the Byron, Unit No. 2, spring 2010 refueling outage. The licensee requested an extension of the CT to 144 hours to replace two of the four SX pump suction isolation valves; maintenance history has shown that replacement of the SX pump suction isolation valves cannot be assured within the existing 72 hour CT window.

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,

Marshall J. David

Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454 and STN 50-455

Enclosures:
1. Amendment No. 168 to NPF-37
2. Amendment No. 168 to NPF-66
3. Safety Evaluation

cc w/encls: Distribution via ListServ

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
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 September 24, 2009, as supplemented by letters dated November 13, 2009, January 19, 2010, March 1, 2010, March 9, 2010 (two letters), and March 19, 2010, 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.168 , 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

Nicholas J. DiFrancesco for

Stephen J. Campbell, 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: April 9, 2010

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
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 September 24, 2009, as supplemented by letters dated November 13, 2009, January 19, 2010, March 1, 2010, March 9, 2010 (two letters), and March 19, 2010, 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.168 . and the Environmental Protection Plan contained in Appendix B, both of which are 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

for

Stephen J. Campbell, 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: April 9, 2010

ATTACHMENT TO LICENSE AMENDMENT NOS. 168 AND 168

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

Remove	Insert
License NPF-37	License NPF-37
License Page 3	License Page 3
License NPF-66	License NPF-66
License Page 3	License Page 3
TSs	TSs
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2

(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. 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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 168, 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 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) 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 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. 168, and the Environmental Protection Plan contained in Appendix B, both of which are 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.

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water (SX) System

LCO 3.7.8 The following SX trains shall be OPERABLE:

a. Two unit-specific SX trains; and

b. One opposite-unit SX train for unit-specific support.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE-------- Not applicable to Unit 1 during replacement of the SX suction isolation valves (i.e., 1SX001A and 2SX001A) during Unit 2 Refueling 15 while Unit 2 is in MODE 5, 6, or defueled. --------------------- One unit-specific SX train inoperable.	A.1 --------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for Emergency Diesel Generator made inoperable by SX. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for Residual Heat Removal loops made inoperable by SX. --------------------- Restore unit-specific SX train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Only applicable to Unit 1 during replacement of the SX suction isolation valves (i.e., 1SX001A and 2SX001A) during Unit 2 Refueling 15 while Unit 2 is in MODE 5, 6, or defueled. ------------------- One unit-specific SX train inoperable.	B.1	--------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for Emergency Diesel Generator made inoperable by SX. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for Residual Heat Removal loops made inoperable by SX. -------------------- Restore unit-specific SX train to OPERABLE status.	144 hours
C. Opposite-unit SX train inoperable.	C.1	Restore opposite-unit SX train to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition A, B or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. NPF-37

AND AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. NPF-66

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

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

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated September 24, 2009 (Reference 7.1), as supplemented by letters dated November 13, 2009 (Reference 7.2), January 19, 2010, (Reference 7.3), March 1, 2010 (Reference 7.4), March 9, 2010 (two letters, References 7.5 and 7.6), and March 19, 2010 (Reference 7.7), Exelon Generation Company, LLC (EGC, the licensee) requested an amendment to Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station (Byron), Unit Nos. 1 and 2. The license amendment request (LAR) is for Technical Specification (TS) 3.7.8, "Essential Service Water (SX) System." The Limiting Condition for Operation (LCO) for this TS states, "The following SX trains shall be OPERABLE: a. Two unit-specific SX trains; and b. One opposite-unit SX train for unit-specific support," and is applicable in MODES 1, 2, 3, and 4. The LAR would create a new Condition B for TS 3.7.8, which would apply to Unit 1, the operating unit, when one of its SX trains is inoperable during the Unit 2 spring 2010 refueling outage, when Unit 2 is in MODE 5, 6, or defueled. The new Condition B effectively allows an increase in the Completion Time (CT) of Required Action A.1, "Restore unit-specific SX train to OPERABLE status," from 72 hours to 144 hours. The existing Condition A would be modified by a note indicating that Condition A is not applicable to Unit 1 during the Unit 2 spring 2010 refueling outage when Unit 2 is in MODE 5, 6, or defueled. Also, TS 3.7.8 would be reformatted to accommodate the addition of new Condition B.

The licensee stated that it requested the changes in order to replace the 1A and 2A SX pump suction valves used for pump isolation from the SX water supply. Currently, the suction isolation valve for the 1A SX pump is degraded such that individual pump isolation may not be adequate to perform pump maintenance or downstream system component maintenance. The same condition is expected for the suction isolation valve for the 2A SX pump. In order to replace these suction isolation valves, the common upstream suction isolation valve for the 1A and 2A SX pumps must be closed and the suction header drained. After draining the common suction header, both the 1A and 2A SX suction isolation valves will be replaced. This evolution is time consuming, and station maintenance history has shown that completion of the needed SX suction isolation valve replacement cannot be assured within the existing 72-hour CT window.

The licensee plans to replace the 1A and 2A SX suction isolation valves during the Unit 2 spring 2010 refueling outage. Closing the common suction isolation valve for the 1A and 2A SX pumps renders one SX train for the operating unit, Unit 1, and one SX train for the shutdown unit, Unit 2, inoperable, and puts Unit 1 into a 72-hour CT for Required Action A.1 under the current TS. Not being able to complete the suction isolation valve replacement in the 72-hour CT would result in Unit 1 being shut down, or in not completing this work to improve the material condition of the plant.

The supplemental letters, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change NRC staff's original proposed no significant hazards consideration determination published in a biweekly notice in the *Federal Register* on December 1, 2009 (74 FR 62835).

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 structures, systems and components important to safety during normal operating and accident conditions.

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, valves, and instrumentation. The pumps and motor-operated valves are aligned by the plant operators, 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 under voltage 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 (AFW) 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 train in each unit is supplied by a single pump rated at 24,000 gpm at 180 feet $\pm$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." The suction supply is one 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. Therefore, the system meets the single-failure criterion as shown in the analysis in UFSAR Tables 9.2-2, "Single Failure Analysis of the Essential Service Water System," and 9.2-16, "Single Failure Analysis of the Ultimate Heat Sink." 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 train 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 Byron, 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, "Ultimate Heat Sink (Byron)."

The design basis of the SX system is for one SX train, in conjunction with the component cooling water (CCW) system and a 100 percent capacity containment cooling system, to remove core decay heat following a design-basis LOCA, as discussed in 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 (ECCS) 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 CCW system, also cools the unit from residual heat removal (RHR) entry conditions, as discussed in UFSAR Section 5.4.7, "Residual Heat Removal System," to Mode 5 during normal and post-accident operations. The time needed for this evolution is a function of the number of CCW and RHR System trains that are operating.

2.2 Applicable Regulatory Requirements and Review Criteria

NRC requirements and review criteria that the NRC staff considered include:

- Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," paragraph (c) (2) (ii) (C), Criterion 3, which requires that a 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 a design basis accident [DBA] or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

- The regulation at 10 CFR, Part 50, Appendix A, General Design Criterion (GDC) 44, "Cooling water," which states in part, "Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. (The NRC staff notes that UFSAR Section 3.1, CONFORMANCE WITH NRC GENERAL

DESIGN CRITERIA, states, in part, "... these stations fully satisfy and are in compliance with the NRC General Design Criteria.")

- 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," Revision 1, November 2002, describes a risk-informed approach, acceptable to the NRC staff, for assessing the nature and impact of proposed licensing-basis changes by considering engineering issues and applying risk insights. This RG provides the key principles for evaluating such changes:

  1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

  2. The proposed change is consistent with the defense-in-depth philosophy. (The RG provides the elements of the NRC staff's defense-in-depth philosophy.)

  3. The proposed change maintains sufficient safety margins.

  4. When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

  5. The impact of the proposed change should be monitored using performance measurement strategies.

  RG 1.174 is directly applicable to long-term (as opposed to single-outage) changes to TS requirements. However, the NRC staff has previously consulted this RG in making risk-informed decisions about single-outage TS changes. Therefore, the NRC staff used the criteria to guide its review and applied them to the extent possible.

- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 0, August 1998, describes a risk-informed approach, acceptable to the NRC staff, specifically for assessing proposed changes in allowed TS outage times (AOTs). Note that the term "CT" used in the licensee's LAR and TSs is equivalent to the term "AOT" used in RG 1.177. This RG provides the elements of the NRC staff's defense-in-depth philosophy. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed CT TS change, as discussed below:

  o Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP).

Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past related applications or additional applications under review are also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage 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 purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 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 appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by 10 CFR 50.65, the "Maintenance Rule," section 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for the application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

For TS changes which implement single-outage requirements, examination of the risk metrics identified in RG 1.174 and RG 1.177 provides insight about the potential risk impacts, even though neither of these RGs provides numerical risk acceptance guidelines for evaluating such TS changes against the fourth key principle (discussed above) of risk-informed applications. It can be demonstrated with reasonable assurance that a TS change implementing long-term requirements meets the fourth key principle, if the associated risk metrics:

- Satisfy the risk acceptance guidelines in RG 1.174 and RG 1.177, or

  - Are not substantially above the risk acceptance guidelines in RG 1.174 and RG 1.177, and effective compensatory measures to maintain lower risk are implemented while the single-outage TS change is in effect.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009, describes an acceptable approach for determining whether the quality of the PRA, in total or the parts used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors.

- General guidance for evaluating the technical basis for proposed risk-informed changes is provided in NRC Standard Review Plan (SRP) Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance." Guidance on evaluating PRA technical adequacy is provided in SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," which includes CT changes as part of risk-informed decision making. Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the five key principles of risk-informed decision making detailed in RG 1.174.

  The NRC staff recognizes that the proposed change to the SX CT is a single-outage change and that the SRP principles are for long-term changes. Nevertheless, for single-outage changes, SRP Sections 19.2 and 16.1 are used to provide general guidance regarding evaluation of the potential risk impacts.

- The NRC staff reviewed the licensee's planned use of additional, dedicated operators to establish reasonable assurance for allowing credit for the proposed actions. For this part of the review, the NRC staff used the guidance contained in NRC Information Notice (IN) 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times;" ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions;" and NUREG-1764, "Guidance for the Review of Changes to Human Actions." Also, the NRC staff used the specific review criteria contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

3.0 TECHNICAL EVALUATION

As stated previously, the LAR would create a new Condition B for TS 3.7.8, which would apply to Unit 1, the operating unit, when one of its SX trains is inoperable during the Unit 2 spring 2010 refueling outage, when Unit 2 is in MODE 5, 6, or defueled. The new Condition B effectively allows an increase in the CT for restoring a unit-specific SX train to OPERABLE status from 72 hours to 144 hours.

In order to accommodate new Condition B, the licensee proposed other changes to TS 3.7.8. The existing Condition A would be modified by a note indicating that Condition A is not applicable to Unit 1 during the Unit 2 spring 2010 refueling outage when Unit 2 is in MODE 5, 6, or defueled. Existing Conditions B and C would be re-lettered to Conditions C and D, respectively. The wording of existing Condition C would be revised in re-lettered Condition D to read "... Completion Time of Condition A, *B or C not met* [change in italics]." The NRC staff finds that these other changes are editorial in nature, preserve the existing links between this TS and other TSs, and are similar to editorial changes previously approved by the NRC staff for a previous Byron SX CT extension (March 18, 2004, Agencywide Documents Access and Management System (ADAMS) Accession No. ML040610869). Therefore, the NRC staff finds these other changes acceptable.

The NRC staff's evaluation of the Unit 1 CT increase to 144 hours for the Unit 2 spring 2010 refueling outage, when Unit 2 is in MODE 5, 6, or defueled, is provided below in the remainder of this safety evaluation.

3.1 Defense-In-Depth

The elements of the defense-in-depth philosophy are described in RG 1.174 and RG 1.177. Table 1 of Attachment 5 of Reference 7.1 provides a summary of the licensee's defense-in-depth assessment of the planned SX configuration; and Table 2 of Attachment 5 of Reference 7.1 provides an evaluation of the RG 1.174 defense-in-depth considerations for the proposed extension of the TS 3.7.8 CT.

3.1.1 Defense-In-Depth Evaluation

The NRC staff's evaluation of each element of the defense-in-depth philosophy follows. Consistency with defense-in-depth philosophy is maintained if:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

  The SX system has safety-related functions in preventing core damage and containment failure by transferring core decay heat and reactor coolant energy during a DBA through the RHR system and the CCW system to the ultimate heat sink. The SX system also provides cooling water to the emergency diesel generators (EDGs), the cubicle coolers and/or oil coolers of the ECCS and AFW pumps, and cooling water to the reactor containment fan coolers (RCFCs). SX also has safety-related functions in accident consequence mitigation by providing cooling water to the cubicle coolers of the containment spray pumps and cooling water to the control room refrigeration units. These SX safety functions are preserved during the extended CT of this LAR, except that the supply of SX cooling water from the 1A SX pump will not be available as a redundant supply. This condition is currently allowed by TS 3.7.8 for 72 hours, whereas, this LAR requests a CT of 144 hours.

  The change to TS 3.7.8 does not introduce a new accident or transient, because no new equipment is installed, existing equipment is not operated in a new manner, and thus no new accident initiator is introduced. The SX system is not an initiator of any analyzed DBA, therefore, the change to TS 3.7.8 does not increase the likelihood of an accident or transient.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

  The LAR does not change the plant design. Since the existing SX system plant design does not permit the planned maintenance on 1SX001A without affecting the operability of 2SX001A and vice versa, both valves will be simultaneously inoperable during the planned maintenance. A protected equipment list will be in effect and additional dedicated operators will be stationed as compensatory actions. The NRC staff does not consider this an over-reliance on programmatic activities. No specific programs are being initiated or relied upon to compensate for the lack of a redundant SX pump and supply line during the extended CT.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency of challenges to the system.

  SX system redundancy will be reduced during the extended CT when the 1A SX train is inoperable. The current allowed CT is for 72 hours. In order to extend the CT to 144 hours during valve replacement, the licensee is relying upon certain plant design and operational features and restrictions which are discussed as follows.

  The licensee will invoke additional precautions to compensate for extending the CT to include protecting equipment important to safety. During the extended CT, maintenance will not be allowed on protected equipment. The protected equipment is listed in Table 3 of Attachment 1 and Table 14 of Attachment 5 of Reference 7.1. Additionally, in its March 1, 2010 (Reference 7.4), response to the NRC staff's request for additional information (RAI) dated February 19, 2010 (Reference 7.9), the licensee committed to protect the safety-related SX cooling water loads of the 1B SX train.

  Although redundancy of the SX system supply is not maintained during the extended CT, an element of redundancy is established in that the licensee states in its letters dated March 1, 2010 (Reference 7.4) and March 9, 2010 (Reference 7.5), that both Unit 1 trains of SX loads are operable during the extended CT with the supply of only one SX pump (1B SX pump). The licensee bases the operability assessment on their safety-related evaluations and SX flow testing, which show that, with the SX trains cross-tied, a single SX pump is capable of providing adequate cooling water to both trains of SX loads under normal and accident conditions. Because Unit 1 will be in proposed new LCO Condition B during the extended CT, the single failure criterion is temporarily relaxed when assessing regulatory compliance, as explained in Generic Letter 80-30, "Clarification Of the Term Operable As it Applies To Single Failure Criterion For Safety Systems Required By TS," dated April 10, 1980. Therefore, an additional failure of the only operable Unit 1 SX pump is not assumed during the extended CT, which supports the licensee's operability assessment of both trains of Unit 1 SX loads only during the extended CT.

  As a compensatory action, the licensee will station additional dedicated operators during the extended CT. The additional dedicated operators will be one Senior Reactor Operator (SRO) in the control room, one Reactor Operator (RO), and one Equipment Operator assigned to monitor SX performance and take the identified actions, if required, as a back up to the nominal shift staff. The only duties assigned to these additional dedicated operators will be those associated with the actions identified in the risk assessment credited to reduce the risk impact. The additional dedicated operators will receive training on their assignments as part of the preparations related to this compensatory measure.

The licensee has procedures in place to respond to a loss of one of the remaining two available SX pumps (1B SX and 2B SX). An NRC staff RAI dated December 18, 2009 (Reference 7.8), asked the licensee to describe the sequence of events and operator actions if an SX pump were lost during the extended CT. In its reply, dated January 19, 2010 (Reference 7.3), the licensee stated that, if the remaining unit-specific SX pump on Unit 1 was lost, established procedures would be followed to cross connect the Unit 2 Train B SX to Unit 1 SX to provide cooling for an immediate shutdown of Unit 1 and cool-down to Mode 5. Temperatures of vital equipment would be monitored; non-essential loads would be shed; and, if necessary, alternate non-safety-related cooling sources established. Furthermore, the licensee stated in its letter dated January 19, 2010 (Reference 7.3), that Byron Abnormal Operating Procedure 0BOA PRI-7, "Loss of Ultimate Heat Sink Unit 0," provides guidance on aligning the fire protection (FP) system to the SX system in event of a complete loss of SX pump capability coincident with an inability to crosstie to the opposite unit SX supply. Alternately, if the lost SX pump is from the shutdown unit (Unit 2) during the extended CT, the SX system for both units would be cross connected and vital equipment monitored and non-essential loads shed. Similarly, in the risk informed evaluation of defense-in-depth in Reference 7.1, the licensee stated that a "best estimate flow analysis" has shown that a single SX pump can provide cooling to both units except the RCFC and EDGs on the unit without an SX pump and one train of RCFCs on the unit with an available SX pump. In its response to a NRC staff RAI dated March 3, 2010 (Reference 7.11), the licensee stated in a letter dated March 9, 2010 (Reference 7.5), that the PRA risk analysis does not take credit for Unit 2 SX supplying Unit 1 EDGs. Therefore, the risk informed evaluation concludes, as described in Table 1 of Attachment 5 of Reference 7.1, that Unit 1 could be placed in hot standby during a total loss of SX to Unit 1, using FP water for reactor coolant pump (RCP) seal cooling and a diesel driven AFW pump or main feedwater for secondary makeup.

An additional compensatory action will be to make available FP water to the chemical and volume control (CV) pumps for RCP seal cooling in the event of an extended loss of SX.

In an RAI dated December 18, 2009 (Reference 7.8), the NRC staff asked the licensee why the TS change (specifically, as discussed in Reference 7.1, Attachment 1, Table 3) does not specify that the removal of 1SX001A and 2SX001A from service will not be scheduled when adverse weather conditions or other situations are predicted that likely may subject the plant to abnormal conditions. In a response dated January 19, 2010 (Reference 7.3), the licensee stated that the extended CT will not be entered should adverse environmental conditions exist or be forecast within the next 12 hours. The license defined adverse environmental conditions as: icing; wind, or storms causing unexpected repeated station power line trips; tornado warning; actual switchyard voltage alarms or notifications indicating voltage below that required for offsite source TS operability limits; predicted unit trip contingency switchyard voltage below minimum required switchyard voltage (unless a site specific analysis has been performed); or notification that at the current time a condition exists such that if a transmission line or other transmission facility were to trip, then site will be below voltage operability limits. The licensee further stated that, should adverse environmental conditions develop after the extended CT condition has been entered, then work will proceed to complete the SX valve replacement. As discussed below in Section 3.3.2, under Evaluation of External Hazards, the risk contribution from these external hazards during the extended CT is acceptable.

- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.

  Defenses against potential common cause failures are preserved, because no new common cause failure mechanisms are introduced by the proposed CT extension.

- Independence of barriers is not degraded.

  Independence of barriers is not degraded, because the proposed CT extension has no impact on the independence of barriers.

- Defenses against human errors are preserved.

  As discussed in Section 3.2, defenses against human errors are preserved.

- The intent of the GDC in Appendix A to 10 CFR Part 50 is maintained.

  The intent of the GDC is maintained, because the proposed CT extension does not modify the design of the plant.

3.1.2 Summary and Conclusion

The safety-related functions of the SX system are preserved during the extended CT, although redundancy of the SX system supply is not available. However, Byron has design features and will invoke compensatory measures to maintain defense-in-depth that is commensurate for the additional 72 hours of the extended CT. Design features include the ability for one unit-specific SX pump (1B SX pump) to supply both unit-specific SX trains (1A and 1B) through the SX train unit-specific crosstie. Also, one SX pump from the shutdown unit (Unit 2) has the ability to supply adequate SX to the operating unit (Unit 1) through the SX opposite-unit crosstie for shutting down Unit 1. In the event of a loss of all SX, the plant has the ability to keep RCP seals cool using FP water to the CV centrifugal charging pumps. Compensatory measures include additional precautions that protect equipment important to safety affected by the 1A SX pump outage. During the extended CT, maintenance activity will not be allowed on protected equipment. Dedicated operators will be stationed to monitor SX system performance and take appropriate action to restore SX flow if needed.

Based on the SX system capabilities, the compensatory actions, and appropriate restrictions as explained and committed to in the licensee's submittals as explained above, the NRC staff finds that the defense-in-depth considerations and compensatory measures committed to by the licensee satisfy the RG 1.174 criteria to the extent practical for a single-outage change to the CT of TS 3.7.8. Because the proposed change will only apply during the Unit 2 spring 2010 outage, the NRC staff considers the defense-in-depth capability credited by the licensee and discussed above to be appropriate, acceptable, and commensurate for the 144 hour CT.

3.2 Operator Performance

As discussed above, Unit 1 (the operating unit) will be in proposed new LCO Condition B during the extended CT, with only train B SX operating, and an additional single-failure, which would fail the Unit 1 train B SX, need not be assumed. Nevertheless, in order to strengthen the level of defense-in-depth and reduce overall risk in accordance with the defense-in-depth philosophy, the licensee included the following proposed manual operator actions to cope with a failure of SX on Unit 1 during the extended CT:

- Establishing an opposite-unit SX crosstie given a failure of the Unit 1 SX
- Establishing alternate cooling paths, such as, FP to CV pump oil cooler given a loss of CCW cooling
- Establishing a cool suction source for CV pumps given a loss of CCW cooling

3.2.1 Staffing Evaluation

In Reference 7.1, the licensee provided information regarding credit for additional operators as a key compensatory measure for the proposed CT extension. The licensee indicated that "dedicated" operators will be utilized in addition to the nominal staffing, i.e., at least, the minimum staffing levels required by the TS and by 10 CFR 50.54, "Conditions of licenses," paragraph (m)(2)(i) plus three additional operators. The NRC staff position (see Safety Evaluation dated March 18, 2004, ADAMS Accession No. ML040610869) regarding "dedicated" is for the "dedicated" 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. Additionally, the term "dedicated" implies that the individual is "qualified" to perform the task.

The licensee indicated that a SRO, a RO, and equipment operator (i.e., non-licensed operator) will be "dedicated" to the three compensatory actions that must be implemented to provide the SX functions if the Unit 1 train B SX fails during the extended CT. The licensee stated that these dedicated personnel will not be assigned any other tasks. These personnel will be located both inside and outside the control room to monitor SX performance and to provide back up to the nominal staff. These personnel will report directly to the operating unit's Unit Supervisor who reports to the Shift Manager. The dedicated equipment operator will report to the dedicated SRO. Standard three-way communication protocol will be used. Communications from the control room to the dedicated equipment operator will be accomplished using the normal radio system, telephone or plant page.

The dedicated SRO and RO will inform the Unit Supervisor if they identify that entry conditions for the compensatory actions have been reached, that the associated abnormal operating procedures have been implemented, and will report the results of the actions. The equipment operator will report to the dedicated SRO.

Because the dedicated operators' duties will supplement nominal unit staff positions, the NRC staff finds the licensee's additional staffing commitment and the command, control, and communication arrangement with the additional dedicated operators to be acceptable.

3.2.2 Operator Action Evaluation

The licensee provided a list of three tasks proposed to be performed by dedicated personnel: establishing the SX unit crosstie, establishing alternate cooling to the CV pump oil cooler, and establishing a cool suction source for CV pumps. The licensee used operator interviews to validate the appropriateness of the proposed operator actions to mitigate a loss of SX initiating event.

The licensee also provided an analysis of the time necessary to perform each of the three tasks, with travel time included as a conservatism. The time required was obtained from a review of operations simulator training records, including job performance measures. Other than the FP connection to the CV pumps, all actions can be taken from the control room. The FP connection to the CV pump is performed in the CV pump rooms in the auxiliary building.

The times required to accomplish the tasks were within the time available. The results reported were:

- Time available to establish SX crosstie given a failure of Unit 1 SX = 90 minutes; time required = 10 minutes.
- Time available to establish alternate cooling, e.g., FP to CV pump oil cooler given loss of CCW cooling = 90 minutes; time required = 20 minutes.
- Time available to establish a cool suction source for CV pumps given a loss of CCW cooling = 30 minutes; time required = 10 minutes.

The NRC staff is satisfied that the proposed operator actions are within the operators' capability to complete within the time constraints used in the analysis.

3.2.3 Preparation and Pre-staging Evaluation

In order to optimize cues to the operator, and to minimize the time and complexity of the compensatory actions, the licensee will do the following:

- Monitor SX pump and system performance (including SX pump motor parameters such as voltage, current, and temperatures).
- Validate the placement of ladders/stepping stools in locations where the FP hose hookup is above the height that an operator can reach without assistance (such as the B charging pump rooms).

In addition, potential sources of human error will be minimized by restricting the amount of work being planned in the work window, especially for licensee-identified "protected" equipment. Generally, work on or within 2 feet of protected equipment will not be allowed.

The NRC staff finds the licensee's proposed preparations to be acceptable because they reduce the number of negative performance-shaping factors involved in the tasks.

3.2.4 Training and Procedures Evaluation

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 LCO 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. The licensee will provide copies of the relevant procedures (i.e., listing the actions to establish the SX unit crosstie, aligning FP cooling for the CV pumps and aligning the CV pump suction from the volume control tank to the reactor water storage tank) to the dedicated control room and equipment operators prior to each shift change to minimize the need to locate the procedures during loss of SX conditions.

The NRC staff finds the licensee's proposed actions regarding training and procedures to be acceptable based on the enhanced immediacy and relevance of the training, and the on-hand availability of the procedures.

3.2.5 Equipment and Environmental Conditions Evaluation

As noted previously, the licensee indicated that ladders/stepping stools will be placed in locations where needed for the designated equipment operator to reach the FP 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 in real-time and have proven to be adequate over the long-term. The licensee stated that, with the exception of the FP 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 based on familiarity with and accessibility of equipment.

3.2.6 Summary and Conclusion

The NRC staff has reviewed the licensee's planned use of additional, dedicated operators as a compensatory measure for extending the CT to restore a unit-specific SX train to operable status from 72 hours to 144 hours. The NRC staff concludes that the licensee has adequately considered the impact of the proposed extension on operator staffing, procedures, equipment, and associated training. The information provided by the licensee establishes reasonable assurance for allowing credit for the proposed actions. Furthermore, based on a technical review of the information provided by the licensee against the guidance in IN 97-78, ANSI/ANS 58.8-1994, NUREG-1764, and SRP Chapters 13 and 18, the NRC staff finds the operator performance aspects of the licensee's LAR acceptable.

3.3 Risk Considerations

3.3.1 Review Methodology

Per SRP Chapter 19 and Section 16.1, the NRC staff reviewed the submittal using the three-tiered approach and the five key principles of risk-informed decision making presented in RG 1.174 and RG 1.177, discussed above in Section 2.2.

3.3.2 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) evaluation of the validity of the Byron PRA models and their application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Quality – Internal Events Model

The objective of the PRA quality review is to determine whether the Byron PRA used in evaluating the proposed changes to TS 3.7.8 CTs is of sufficient scope, level of detail, and technical adequacy for this application. The NRC staff review evaluated the PRA quality information provided by the licensee in their submittals, including industry peer review results and self-assessments performed by the licensee.

The Byron PRA model Revision 6E (June 2009) addresses both CDF and LERF, and is highly detailed, including a wide variety of internal initiating events, modeled systems, operator actions, and common cause events. The model includes a loss of SX initiator which can assess the impact of train unavailability for this application, unit and train crosstie capability, plant-specific data, and operator actions for mitigating a loss of SX initiating event.

The licensee has administrative procedures and processes for configuration control of the PRA model, which includes review of design changes, procedure changes, engineering calculations, and capture of plant-specific data related to equipment performance and initiating events. There are four plant changes identified as not yet incorporated into the Byron PRA model. Two of the changes would reduce the baseline risk, and the other two changes would have only minimal impact on the PRA model.

In 2000, the Pressurized-Water Reactor Owners Group performed a peer review of the Byron PRA model. Since the completion of the peer review, all significant findings and observations have been addressed. In 2008, the licensee also conducted a self-assessment of the Byron PRA against Addendum B of the American Society of Mechanical Engineers (ASME) PRA Standard using RG 1.200, Revision 1, the revision in effect at the time. The results of this self-assessment were identified as "gaps." The licensee provided its summary of the gaps in the

LAR; and the NRC staff evaluated the gaps using RG 1.200, Revision 2. The results of the NRC staff's evaluation follow:

> Gap #1: The summary of the PRA results does not identify significant contributors or describe significant accident sequences or functional failure groups, because an update is not yet complete. This is a documentation issue which does not impact the risk results for this application.
>
> Gap #2: The significance of assumptions used in the PRA model is not assessed. This has been completed for this specific application and is provided in the submittal; therefore, there is no impact on this application.
>
> Gaps #3 and #6: The quantitative definition of "significant" is not documented. This is a documentation issue and does not impact the PRA model results; therefore, there is no impact on this application.
>
> Gaps #4 and #5: The LERF analysis is based on NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," and satisfies capability Category I. Uncertainties have not been characterized formally. This model is conservative, and refinement to capability Category II would not change the conclusions of this application.
>
> Gap #7: The standard requirements for model uncertainties and related model assumptions have been revised. The application-specific evaluation of uncertainties and model assumptions has been completed for this specific application and provided in the submittals; therefore, there is no impact on this application.
>
> Gap #8: Model documentation is not yet updated to reflect recent updates. This task is ongoing, and has no impact on the results supporting this application.
>
> Gap #9: Plant-specific data for motor-operated and air-operated valves is not collected. This application is for unavailability of SX trains, and therefore, such plant data are not significant to this application.

Based on the licensee's dispositions above, the NRC staff agrees that the outstanding gaps would not have a significant impact on the internal events risk assessment results supporting this request. Therefore, the NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), RG 1.200, and SRP Section 19.1, and that the quality of the Byron internal events PRA is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

PRA Quality – Internal Fires Model

The licensee provided a separate quantitative estimate of the impact of the SX train outage on fire risk, which uses a separate fire PRA model based on the methods of NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," as its foundation. The licensee characterized the model as conservative, and used the results to show that a bounding evaluation of fire risk would not change the acceptability of the proposed change based on the quantitative risk criteria. The fire PRA model has the following conservative characteristics:

- Many scenarios assume all equipment and cables within the compartment being evaluated are damaged by the fire, which overstates the extent of damage which would realistically be expected to occur given an actual fire.

- No detailed fire modeling has been applied.

- Recovery actions are not considered for many fire areas.

- Heat release rates and the fire development timelines are conservatively biased.

The licensee identified that the fire PRA was developed using the methodology of NUREG/CR-6850, which is judged to satisfy the key attributes of fire PRAs found in RG 1.200, Section 1.2.4. Specific areas identified by the licensee where the fire PRA methodology used differed from that found in NUREG/CR-6850 are discussed below:

- The fire PRA used available industry guidance on multiple spurious operations due to fire-induced circuit failures. This application focuses on fire-induced failures affecting the Unit 1 train B SX system (while Unit 1 train A is out of service), and spurious operations impacts are not expected to be train specific, and will have similar impacts on both the baseline configuration and the analyzed configuration. Therefore, the potential impact of spurious operations to the change in risk is not expected to be significant.

- Explicit identification and modeling of instrumentation required to support credited operator actions has not been incorporated into the fire PRA. The licensee stated that this would not result in train-specific risk increases affecting this application. Conservatively, local operator actions required within two hours of the initiating event are not credited in the fire PRA, which would reduce the impact of this deficiency. For this specific application, dedicated staff will be in the main control room to monitor affected equipment and take appropriate actions also mitigates this modeling deficiency.

- The failure of the balance-of-plant support system is conservatively assumed when detailed cable routing information is not available. Other components modeled in the PRA for which detailed cable routing information is not available are similarly assumed to be failed, unless a particular area is judged unlikely to include cables for specific systems. These conservatisms do not result in train-specific impacts on the baseline risk or on the configuration-specific risk.

- LERF is not considered in the fire PRA; the impact of fires on large early releases is primarily associated with the potential fire-induced failure of containment boundary components. The SX outage configuration has no unique impact on the containment isolation capability of these components. The internal events LERF results for this application also show a very small relative change compared to the CDF change, and therefore, it is expected that the fire LERF impact would also be very small.

- The scoping fire modeling evaluations did not consider the formation of hot gas layers. The licensee stated that it did not expect hot gas layer formation to be a significant contributor to plant-specific risk due to the large volumes available for dissipation, and provided a detailed assessment of the plant layout to justify this assumption.

- The multi-compartment review has not been completed to confirm that the compartment boundaries are sufficiently robust to prevent inter-compartment spread of fires, or to ensure the analysis of such spread is conducted. The licensee stated that the plant layout makes fire propagation to multiple compartments unlikely, and any such impacts would not be train-specific and so would not significantly impact the application risk analyses.

- A detailed analysis of control room abandonment scenarios has not been performed. The ability to control the operable SX train is not affected since controls for both trains are available at the remote shutdown panels.

- Seismic-fire interaction is not required by NUREG/CR-6850 to be quantified, and any such vulnerabilities would not be train-specific. Therefore, any contribution of seismic-induced fires is expected to be very small for this application.

- Uncertainty and sensitivity analyses have not yet been performed. The conservative nature of the current analysis inputs, such as, ignition frequencies and extent of fire damage, ensures that the overall fire risk estimated is conservative. This conservatism is not train-specific, and so will not impact the application-specific risk analyses.

- The heat release rate specified for evaluating transient fires recommended in Table E-1 of NUREG/CR-6850 was evaluated as overly conservative and was replaced with a more realistic, but still bounding, heat release rate associated with a large electric motor fire. This treatment remains conservative compared to typical transient combustibles, such as extension cords, work lamps, and space heaters.

Although no external reviews were identified for the fire PRA, the licensee stated that internal reviews by engineering staff from other Exelon facilities has been conducted, and the results of these reviews incorporated into the fire PRA.

Based on the general conformance to NUREG/CR-6850 as discussed in the above items, the conservative assumptions applied, and the internal reviews conducted by the licensee, the NRC staff finds for this particular application of the fire PRA that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), RG 1.200, and SRP Section 19.1, and that the quality of the fire PRA and methods applied is sufficient to support the risk evaluation provided by the licensee in support of the proposed license amendment.

PRA Risk Results and Insights

The SX system is modeled in the PRA as impacting both mitigation and initiating events, and therefore, SX outage can be directly modeled in the PRA by assuming the associated pumps are unavailable and unrecoverable. Unavailability of other plant equipment at their nominal average values is assumed, except for the specific equipment identified as assumed to be available by regulatory commitment. Failure modes associated with SX valves failing to open or close, where compensatory measures are identified for specific valve alignments, were eliminated since the valves in question would already be in the required position prior to entering the action requirement. The licensee also credited a reduction in the dependency between operator actions when dedicated Operations personnel are identified for the SX outage.

The ICCDP and the ICLERP are based on the entire 144-hour duration of the proposed extended CT.

The licensee's methodology is consistent with the guidance of RG 1.177, Section 2.3.4 and Section 2.4 and is, therefore, acceptable to the NRC staff.

The licensee presented risk results for internal events and for internal fire events. The results are as follows:

Risk Measure	Internal Events	Internal Fires
ICCDP	1E-7	9E-7
ICLERP	3E-9	Not evaluated, assumed not significant

The licensee did not explicitly provide an estimate of the ΔCDF and ΔLERF associated with this proposed change. The ΔCDF and ΔLERF are determined by assuming a frequency for entry into an extended CT of this nature. Because the proposed TS change is a single refueling outage, it can be conservatively assumed that the frequency of the extended CT is 1/year, and so the ΔCDF and ΔLERF are numerically identical with the ICCDP and ICLERP. Alternatively, the licensee has entered an extended SX outage once before in 2004, so the frequency could be estimated as 2/5 year, which would give a lower estimate of the ΔCDF and ΔLERF.

Per RG 1.177, the acceptance guidelines for ICCDP and ICLERP are 5E-7 and 5E-8, respectively, for permanent changes to the TS. Per RG 1.174, the acceptance guidelines for ΔCDF and ΔLERF are 1E-6/year and 1E-7/year, respectively, for very small changes in risk, also for permanent changes. The licensee's cumulative estimate internal and fire risk of approximately 1E-6 ICCDP and 3E-9 ICLERP, and the NRC staff's calculation based on 1/year frequency of 1E-6/year ΔCDF and 3E-9/year ΔLERF, are reasonably consistent with these guidelines applicable to permanent changes. Therefore, the NRC staff finds that the risk is acceptable to permit the requested CT extension for a single refueling outage.

Evaluation of Seismic Risk

The licensee did not quantitatively assess the impact of seismic events on CDF or LERF, but instead provided a qualitative assessment of the impact of seismic events during the SX outage. The risk impact is expected to be insignificant based on the following:

- The plant configuration during the service water outage maintains the success path of two available pumps, and has no impact on the seismic capability of the SX system.

- The frequency of a seismically-induced loss of offsite power is estimated at 5E-5/year, and the probability of a SX pump failing to start and run is approximately 1E-3; therefore, over the 144-hour CT, the additional risk contribution from seismically-induced station blackout (due to loss of cooling to the associated EDG) is about (5E-5/yr)(1E-3)(144hr)/(8760hr/yr) = 8E-10. This is more than three orders of magnitude less than the other quantified sources of configuration risk, and is not significant.

- The SX system is a seismically qualified backup source for the AFW pumps. However, a seismically-induced failure of the SX pump would fail the EDG and thus directly fail the power source for the AFW pump, and so the unavailability of the SX pump for this function has no additional impact on seismic risk.

- The SX system is a seismically qualified backup source for the fire protection system. The Byron individual plant examination of external events (IPEEE) investigated the potential for seismically-induced fires and found no significant source of risk.

Based on these considerations, the NRC staff concludes that seismic risk is not a significant contributor to the risk during the SX outage.

Evaluation of External Hazards

The licensee stated that Byron conforms to the SRP, June 1987, with regards to external hazards, as discussed in the IPEEE submittal. (Chapter 2 of the SRP provides evaluation criteria for external hazards.) Therefore, no vulnerabilities to other external events exist at Byron. Furthermore, as an extra measure of defense-in-depth, the licensee stated that the extended CT will not be entered should adverse environmental conditions exist or be forecast within the next 12 hours. Nevertheless, the licensee evaluated the external hazards considered in the SRP, and identified three initiators which could be potentially affected by the SX outage.

- Severe Temperature Transients – very low or very high temperatures could impact the functionality of the SX system. Refueling outages are scheduled generally in the spring or fall (in the spring of 2010 for this LAR) when such temperature transients are unlikely.

- Severe Weather Storms – high winds and tornadoes would not affect the SX system since the pumps and all associated support systems are within buildings designed to withstand the affects of high winds. These initiators cause a loss of offsite power, and such events are already considered in the internal events PRA data.

- Lightning Strikes – these can cause a loss of offsite power, and such events are already considered in the internal events PRA data.

Based on the conformance of the plant to the SRP and the evaluation of specific events which are relevant to the SX outage, the NRC staff concludes that the risk contribution from other external hazards is not significant for this application.

Shutdown Risk

The licensee's submittal did not specifically address shutdown risk in the Tier 1 risk evaluation, since the proposed change to the TS has no impact for the shutdown unit. The licensee did provide its evaluation of the impact on the shutdown unit qualitatively in the Tier 2 evaluation, discussed in Reference 7.1, Section 2.3.2.2.

Uncertainty Analysis

The licensee stated that its PRA model does not currently have the capability to quantitatively characterize parametric uncertainty. Instead, a review of cutsets was performed to identify cutsets with basic events whose failure probabilities are derived from the same parameters, and therefore subject to the effect of epistemic uncertainties. The review for this application identified that the 2.7 percent of the internal events ΔCDF and 1.93 percent of the ΔLERF are from cutsets with correlated basic events, and the highest frequency cutset subject to the epistemic uncertainty occurs at the 50th CDF cutset and the 139th LERF cutset. Therefore, the licensee concludes that the potential impact on the application-specific risk results from epistemic uncertainties is not significant, and the use of the point estimates for CDF and LERF is reasonable.

The licensee also evaluated PRA model assumptions to identify those which are key for this application, specifically related to dependency among operator actions, data associated with SX pump failures, and feed-and-bleed cooling assumptions related to the number of power-operated relief valves required for adequate core cooling. The licensee dispositioned these items either by justifying the assumption or providing a reasonable sensitivity study to conclude that any impact on the risk calculation would not be significant. In addition, baseline PRA model sources of uncertainty were similarly reviewed and dispositioned as to the potential for impact on the risk analyses for this application. The licensee also provided sensitivity studies on the compensatory measures associated with this application, and did not identify any particular compensatory measure with high risk significance.

3.3.3 Tier 2: Avoidance of Risk-Significant Plant Configuration

The licensee identified components that would be considered "protected equipment" (as defined by administrative procedures,) which will be required to be available during the SX outage. Protected equipment ensures that no intrusive maintenance activities are performed that would render, or potentially render, the equipment incapable of performing its function. Equipment was selected based on insights from the risk results supporting this application. The licensee provided details from its procedures which provide specific guidance on work activities for protected equipment to clarify the scope of the Tier 2 restrictions. These maintenance restrictions are reflected in the risk analyses supporting this amendment request.

In response to a NRC staff RAI, the licensee clarified the protected equipment scope in its January 19, 2010, response to question 3b (Reference 7.3) and provided an updated commitment in its March 9, 2010, Attachment (Reference 7.6).

The licensee used the results from its fire PRA evaluation of the outage configuration to identify those fire zones in the plant which are the highest fire risk scenarios representing 75 percent of the risk increase for the configuration. Nine fire zones were so identified, and the licensee identified a compensatory measure to perform a walk down of each zone to verify control of transient combustibles prior to entry into the SX outage consistent with the administrative limits of the Byron Fire Protection Program. In response to a NRC staff RAI, the licensee clarified the location of these zones in its January 19, 2010, response to question 3d (Reference 7.3) by qualitatively describing the zones and the important plant equipment in the zones. Further, the licensee enhanced its commitment to require removal of transient combustibles and prohibition of hot work in the proximity of important components in these areas, as stated in its March 9, 2010, updated commitment in Attachment 3 (Reference 7.6).

3.3.4 Tier 3: Risk-Informed Configuration Risk Management

The licensee stated that its CRMP ensures that the risk impact of equipment out of service is appropriately evaluated prior to performing any maintenance activity. The program provides for proceduralized risk-informed assessment of equipment unavailability, including CDF and LERF aspects, and requires assessment for both planned and unplanned activities. Administrative controls are implemented based on the level of risk, including protection of risk-significant equipment and/or expedited equipment restoration.

In its January 19, 2010, response to RAI question 4 (Reference 7.3), the licensee clarified that its Tier 3 evaluation of configuration changes during the SX outage would consider both internal events risk and fire risk.

3.3.5 Summary and Conclusion

The risk impact of the proposed 144-hour CT for the repair of SX valves, as reflected in ΔCDF, ΔLERF, ICCDP, and ICLERP, is consistent with the acceptance guidelines specified in RG 1.174, RG 1.177, and NRC staff guidance outlined in SRP Section 16.1. The Tier 2 evaluation identified the applicable risk-significant plant equipment outage configurations needing compensatory measures that will be implemented by the licensee prior to and during the SX outage. The licensee's CRMP satisfies the CRMP requirements of RG 1.177. Therefore, the NRC staff finds that the risk analysis methodology and approach used by the licensee to estimate the risk impacts and manage configuration risk during the extended CT are reasonable and of sufficient quality.

Based on the above, the NRC staff finds the proposed single-outage change to extend the CT of Required Actions of TS 3.7.8 to be acceptable.

3.4 Current Regulations and Safety Margin Evaluation

Based on the discussions and conclusions provided in Sections 3.1, 3.2, and 3.3 above, the NRC staff finds that the extended CT is not in conflict with any regulatory requirements, codes, or standards relevant to the SX system.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility's components 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 (74 FR 62835; December 1, 2009). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

7.1 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated September 24, 2009 (ADAMS Accession No. ML092680090).

7.2 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Clarification Regarding License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated November 13, 2009 (ADAMS Accession No. ML093200065).

7.3 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Additional Information Supporting License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated January 19, 2010 (ADAMS Accession No. ML100200075).

7.4 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Additional Information Supporting License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated March 1, 2010 (ADAMS Accession No. ML100610109).

7.5 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding the One-Time Extension of the Essential Service Water Train Completion Time," dated March 9, 2010 (ADAMS Accession No. ML100700613).

7.6 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Additional Information Supporting License Amendment Request for a One-Time Extension of the Essential Service Water Train Completion Time," dated March 9, 2010 (ADAMS Accession No. ML100700557).

7.7 Letter from Patrick R. Simpson (Manager - Licensing), Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Clarification to Response to Request for Additional Information Regarding the One-Time Extension of the Essential Service Water Train Completion Time," dated March 19, 2010 (ADAMS Accession No. ML100780401).

7.8 Letter from M. J. David (U. S. Nuclear Regulatory Commission) to C. G. Pardee (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2-Request for Additional Information Related to One-Time Extension of Essential Service Water Train Completion Time (TAC Nos. ME2293 and ME2294)," dated December 18, 2009 (ADAMS Accession No. ML093200660).

7.9 Letter from M. J. David (U. S. Nuclear Regulatory Commission) to C. G. Pardee (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2-Request for Additional Information Related to One-Time Extension of Essential Service Water Train Completion Time (TAC Nos. ME2293 and ME2294)," dated February 19, 2010 (ADAMS Accession No. ML100491351).

7.10 E-mail from M. J. David (U. S. Nuclear Regulatory Commission) to P. R. Simpson (Exelon Generation Company, LLC), " BYRON 1 & 2 - RAI FOR ONE-TIME EXTENSION OF ESW TRAIN COMPLETION TIME (ME2293-94)," dated February 26, 2010 (ADAMS Accession No. ML100600914).

7.11 E-mail from M. J. David (U. S. Nuclear Regulatory Commission) to P. R. Simpson (Exelon Generation Company, LLC), "BYRON 1 & 2 - RAI FOR ONE-TIME EXTENSION OF ESW TRAIN COMPLETION TIME (ME2293-94)," dated March 3, 2010 (ADAMS Accession No. ML100630792).

Principal Contributors: J. Purciarello, NRR
A. Howe, NRR
G. Lapinsky, NRR

Date: April 9, 2010

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

Sincerely,

/RA/

Marshall J. David, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454 and STN 50-455

Enclosures:
1. Amendment No. 168 to NPF-37
2. Amendment No. 168 to NPF-66
3. Safety Evaluation

cc w/encls: Distribution via ListServ

DISTRIBUTION:
PUBLIC
LPL3-2 R/F
RidsNrrDorlLpl3-2 Resource
RidsNrrDorlDpr Resource
RidsNrrLATHarris Resource
RidsNrrPMByron Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn3MailCenter Resource
RidsNrrDssSbpb Resource
RidsNrrDraApla Resource
RidsNrrDirsIhpb Resource
RidsNrrDeEeeb Resource
RidsNrrDirsItsb Resource
RidsOgcRp Resource
JPurciarello, NRR
AHowe, NRR
GLapinsky, NRR
MHamm, NRR
KMiller, NRR

ADAMS Accession No.: ML100740224 *SE date NRR-058

OFFICE	LPL3-2/PM	LPL3-2/LA	DSS/SBPB/BC	DRA/APLA/BC	DIRS/IHPB/BC	DE/EEEB/BC	DIRS/ITSB/BC	OGC	LPL3-2/BC
NAME	MDavid	THarris	GCasto*	DHarrison*	UShoop*	GWilson	RElliott	DRoth	SCampbell NDiFrancesco for
DATE	3 / 29 /10	3/ 26/10	03/12/10	03/17/10	03/15/10	03/ 15/10	03/ 29/10	4/08/10	4/ 9/10

OFFICIAL RECORD COPY