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

November 23, 2016

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNIT 2 - ISSUANCE OF AMENDMENTS REGARDING 2A ESSENTIAL SERVICE WATER PUMP TECHNICAL SPECIFICATIONS FOR PUMP REPAIR (CAC NO. MF8438)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 191 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Unit No. 2. The amendment is in response to your application dated September 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A474), as supplemented by letters dated October 26 and 28, 2016, and November 14, 2016 (ADAMS Accession Nos. ML16301A073, ML16302A468, and ML16319A397).

The amendment adds a Required Action A.2 that increases the completion time currently specified in Required Action A.1, "Restore unit-specific SX train to OPERABLE status," associated with Technical Specifications (TS) Section 3.7.8, "Essential Service Water (SX) System," from 72 hours to 200 hours. This proposed change will only be used one time during a planned 2A SX pump repair

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,

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-457

Enclosures:
1. Amendment No. 191 to NPF-77
2. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 191
Renewed 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 September 30, 2016, as supplemented by letter(s) dated October 26, October 28, and November 14, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 191 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed 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 prior to the 2A SX pump work window.

FOR THE NUCLEAR REGULATORY COMMISSION

G. Edward Miller, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: November 23, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 191

RENEWED FACILITY OPERATING LICENSE NO. AND NPF-77

DOCKET NO. STN 50-457

Replace the following pages of the Renewed 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- 77	License NPF-77
Page 3	Page 3
TSs	TSs
Page 3.7.8-1	Page 3.7.8-1
Page 3.7.8-2	Page 3.7.8-2
	Page 3.7.8-3
	Page 3.7.8-4

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

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

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

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

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3645 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 191 and the Environmental Protection Plan contained in Appendix B, both of which are attached to Renewed License No. NPF-72, dated January 27, 2016, are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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. One unit-specific SX train inoperable.	------------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. -----------------------------	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	--------NOTE-------- Not applicable to Unit 2 during repair of the 2A SX pump during the one-time Unit 2 planned SX System outage to be completed no later than January 23, 2017. --------------------	
		Restore unit-specific SX train to OPERABLE status.	72 hours
	<u>OR</u>		
	A.2	--------NOTE-------- Applicable to Unit 2 during repair of the 2A SX pump during the one-time planned SX System outage to be completed no later than January 23, 2017. Allowance of the extended completion time is contingent on meeting the compensatory measures described in EGC submittal letter RS-16-197. --------------------	
		Restore unit-specific SX train to OPERABLE status.	200 hours
B. Opposite-unit SX train inoperable.	B.1	Restore opposite-unit SX train to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	-------------------NOTE-------------------- Isolation of SX flow to individual components does not render the SX System inoperable. --- Verify each unit-specific SX manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	-------------------NOTE-------------------- Not required when opposite unit is in MODE 1, 2, 3, or 4. --- Operate the opposite-unit SX pump for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Cycle each opposite-unit SX crosstie valve that is not secured in the open position with power removed.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.8.4	Verify each unit-specific SX automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.5	Verify each unit-specific SX pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191 TO RENEWED FACILITY

OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNIT 2

DOCKET NO. 50-457

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated September 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A474), as supplemented by letters dated October 26, 2016 (ADAMS Accession No. ML16301A073), October 28, 2016 (ADAMS Accession No. ML16302A468) and November 14, 2016 (ADAMS Accession No. ML16319A397), Exelon Generation Company, LLC (the licensee) requested changes to the technical specifications (TSs) and facility operating license for the Braidwood Station, Unit 2 (Braidwood). The proposed change modifies TS 3.7.8 by adding a new Required Action A.2 that increases the completion time (CT) currently specified in Required Action A.1, "Restore unit-specific SX [emergency service water] train to OPERABLE status," from 72 hours to 200 hours. The proposed change is a one-time change to support a planned 2A SX pump repair scheduled to be performed before January 23, 2017.

The October 26 and 28, 2016, and November 14, 2016, supplements contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The SX system removes 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 nonsafety-related components.

The SX system consists of two separate, electrically independent, 100 percent capacity for safety-related and cooling water trains. Each train consists of a 100 percent capacity pump, piping, valves, and instrumentation. The pumps and valves are remotely and manually aligned. and are automatically started upon receipt of a safety injection signal or an undervoltage on the engineered safety function (ESF) bus, and all essential valves are aligned to their required accident positions (diesel generator (DG)) system is the backup water supply to the auxiliary feedwater (AFW) system.

The SX system includes provisions to crosstie the trains (unit-specific crosstie), as well as provisions to crosstie the units (opposite-unit crosstie). The opposite-unit crosstie valves (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.

The SX system provides cooling water for: (1) DG coolers, (2) containment fan coolers, (3) component cooling heat exchangers, (4) diesel- and motor-driven AFW pump lube oil coolers, (5) diesel-driven AFW pump cubicle coolers, (6) diesel-driven AFW pump coolers, (7) SX pump lube oil coolers, (8) SX pump cubicle coolers, (9) centrifugal charging pump cubicle coolers, (10) centrifugal charging pump oil coolers, (11) positive displacement charging pump cubicle cooler, (!2) safety injection pump cubicle coolers, (13) safety injection pump oil coolers, (14) containment spray pump cubicle coolers, (5) residual heat removal (RHR) pump cubicle coolers, (6) control room refrigeration units, (7) spent fuel pit pump cubicle coolers, and (18) primary containment coolers.

2.0 REGULATORY EVALUATION

The NRC staff finds that the licensee, in Section 4.1 of its submittal, identified the applicable regulatory requirements for which the NRC staff based its acceptance are listed below.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, TSs, Paragraph (c) (2) (ii) (C), Criterion 3, which requires that TS limiting condition for operation (LCO) be established for: A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate the DBA 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 Decision making: Technical Specifications," contains five key principles of the NRC staff's philosophy of risk-informed decision making. 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 (DID) 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 NRC 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.

SRP, Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," provides general guidance for evaluating the technical basis for proposed risk-informed changes. Guidance on evaluating probabilistic risk assessment (PRA) technical adequacy is provided in SRP, Chapter 19, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load."

Regulatory Guide (RG) 1.174, Revision 2, "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 permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" May 2011. (ADAMS Accession No. ML100910008), describes an acceptable risk-informed approach specifically for assessing proposed one-time TS changes in CTs. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 provides the following three-tiered TS acceptance guidelines specific to one-time only CT changes for evaluating the risk associated with the revised CT:

1. The licensee has demonstrated that implementation of the one-time only TS CT change impact on plant risk is acceptable (Tier 1):

    - Incremental conditional core damage probability (ICCDP) of less than 1.0×10^{-6} and an incremental conditional large early release probability (ICLERP) of less than 1.0×10^{-7}, or

    - ICCDP of less than 1.0×10^{-5} and an ICLERP of less than 1.0×10^{-6} with effective compensatory measures implemented to reduce the sources of increased risk.

2. The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2).

3. The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program (Tier 3).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009. (ADAMS Accession No. ML090410014) describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors.

SRP Chapter 18, "Human Factors Engineering," Revision 2

NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3

NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1

3.0 TECHNICAL EVALUATION

3.1 Evaluation of Key Risk Principles

RG 1.177 provides the following five key principles for making risk-informed decisions about proposed TS changes:

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 DID philosophy.
3. The proposed change maintains sufficient safety margins.
4. When the 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.
5. The impact of the proposed changes should be monitored using performance measurement strategies.

The first three principles pertain to traditional engineering considerations and the last two principles involve risk considerations.

3.1.1 Key Principle 1: Compliance with current regulations

The regulations pertinent to the licensee's proposed TS amendment request are 10 CFR 50.36(c) (2) LCO, (i) and (ii), Criterion 3 and 4. The licensee's proposed a one-time change to TS 3.7.8 to increase the CT for an inoperable SX train affects the maximum allowed time to have only one SX train OPERABLE without shutting down the reactor. The requested change does not propose any deviation or exemption to the regulation itself, but only a one-time change to how the regulation is implemented. Therefore, the proposed change is in compliance with current regulations.

3.1.2 Key Principle 2: Defense in depth

RG 1.177 states that consistency with DID philosophy is maintained if the following seven items are met:

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

   Prevention of core damage depends on the ability to continuously remove decay heat after an initiating event. During the extended CT of 200 hours, if a loss of offsite power (LOOP) occurred, the 2B SX train remains available to mitigate the event. If a failure occurred in the 2B EDG causing failure of the 2B SX train, the licensee is able to cross connect SX from Unit 1 to supply sufficient SX to remove decay heat and cool containment as explained below regarding system redundancy, independence and diversity. The license amendment request (LAR) does not adversely affect consequence mitigation features because only the 2A SX train is affected. Therefore, the NRC staff considers that there is a reasonable balance among prevention of core damage, prevention of containment failure and consequence mitigation.

2. Over reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.

   The licensee's PRA identified that compensatory actions would be needed to account for fire-based ICCDP greater than 1.0E-06 but less than 1.0E-05. The licensee also provided compensatory action for internal events. The compensatory actions are listed in Exelon LAR letter RS-16-197(ML16274A474), dated September 30, 2016, Attachment, 1 pages 11 through 13. In response to an email request for additional information (RAI) dated October 20, 2016, which asked the licensee whether the compensatory actions were an overreliance on programmatic activities, the licensee in a letter dated October 26, 2016 (ADAMS Accession No. ML16301A073) stated:

   > The compensatory actions specified address elimination of elective maintenance, staff briefings, and plant walk downs to ensure the plant configuration is consistent with the risk analysis. None of the compensatory actions are new or revised actions for the plant staff. These types of actions are not complex, are controlled through plant procedures and programs, and are frequently performed. The fire related compensatory actions are focused on planned activities to prevent a fire event rather than emergent activities focused on mitigating a fire event; therefore, they are reliable and will be effectively executed.

   The NRC staff finds that these actions do not represent an overreliance on programmatic activities for this one-time TS change because the compensating actions are in addition to other defense in depth attributes listed here and aren't the sole method of achieving DID.

3. System redundancy, independence, and diversity are maintained commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

   System redundancy, independence, and diversity are normally achieved with two SX trains per unit. During the repair of the 2A SX pump, the 2B SX train will be the only OPERABLE SX train in Unit 2. The 2B SX train can mitigate all DBAs provided there are no failures in the 2B SX train. The NRC staff considered the combined probability of a DBA loss-of-coolant accident (LOCA) event with a failure of the 2B SX train to be extremely small during the extended CT and, thus, not require additional DID compensatory action for LOCA mitigation. However, the combined probability of a LOOP and failure of an emergency power source should be considered as a DID consideration regarding redundancy. Therefore, the NRC staff in an RAI dated October 20, 2016 (ADAMS Accession No. ML16294A343), asked the licensee to address the scenario of an extended LOOP with a failure of the 2B EDG. In the RAI, the NRC staff asked for operator action and response times, method of primary pressure and inventory control, steam generator (SG) feed supply sources for an extended LOOP, and whether analyses were in place to demonstrate whether the cross flow from Unit 1 is sufficient. The licensee responded in a letter dated October 26, 2016 (ADAMS Accession No. ML16301A073), and stated that there are procedures in place to cross- connect SX from Unit 1 from the control room to supply Unit 2 SX loads with the standby SX pump from

Unit 1. Simulator training and demonstrations have shown that this evolution can be completed within 5 minutes. The licensee stated that the Unit 2 EDGs were designed to operate without cooling water for 5 minutes and an engineering evaluation by the licensee determined that an EDG could operate for up to 22 minutes without cooling water. Therefore, electrical power would be restored to Unit 2 as soon as a Unit 2 EDG started (within 10 seconds of the LOOP) and be able to continuously run. The energization of the Unit 2 essential buses after the LOOP provide sufficient power to pressurizer heaters, charging (CV) pumps, letdown and component cooling (CC) pumps to control primary pressure and inventory, provide auxiliary feed water to the steam generators for decay heat removal, power RHR and CC pumps for shutdown cooling for cold shutdown and decay heat removal if required, and power CV and CC pumps for reactor coolant pump (RCP) seal injection and cooling. The containment would be cooled by one train of reactor containment fan coolers (RCFC) soon after SX is restored to Unit 2. Previous station blackout analysis demonstrate that the containment temperature will remain below equipment qualification limits for nearly 4 hours without SX to the RCFCs. Therefore, containment temperature limits would not be approached when mitigating a total loss of SX and using SX from the cross connect between units. The licensee further stated that the updated final safety analysis report (UFSAR). Section 9.2.1.2, states that as soon as the SX crosstie has been completed, the Unit 1 SX system supplies all required SX loads for both units with the two SX pumps from Unit 1. Additionally, the licensee will invoke the compensatory measures identified in the LAR letter RS-16-197 dated September 30, 2016, Attachment 1, pages 11 through 13. The NRC staff finds the licensee's response satisfactory in that they have sufficient redundancy and compensatory measures in place to mitigate a LOOP and failure of the remaining SX train for this one-time CT extension.

4. Defenses against potential common cause failures (CCF) are maintained and the potential for introduction of new CCF mechanisms is assessed.

   The replacement of the rotating element of the 2A SX pump is preemptive. The pump has not failed. The cause of the degrading performance is expected wear and consistent with the operating history of all the Braidwood SX pumps. Typically, the SX pump repair is needed about every 15 years because of operating wear. The average differential pressure across each SX pump, which indicates when pump repair is needed, is identified in the LAR. Both SX pumps 1B and 2A are in the differential pressure (DP) Alert Range. SX pump 1B rotating element was replaced in the October 2016, outage. The other SX pumps, 1A and 2B, have DP values in the normal range demonstrating that the need for repair is not needed in the near future. Since failure of the other SX pumps due to wear is not imminent, common cause failure is not credible.

5. Independence of physical barriers is not degraded

   With the ability to continuously remove decay heat, the proposed change does not affect the fuel cladding. The RCP boundary and containment cooling are not challenged by this LAR. Therefore, the independence and integrity of independent barriers are not challenged.

6. Defense against human errors is maintained

   The licensee has identified briefings to be held by Operations for the following actions: (1) loss of RCP seal cooling, (2) loss of AFW, (3) loss of Unit 2 SX, (4) refilling the diesel-driven AFW pump day tank, 5) action on loss of a vital instrument bus. Shift briefings prior to entering TS 3.7.8 Action Statement and throughout the CT period will also be conducted. The licensee has procedures in place to direct cooling of Unit 2 SX loads using Unit 1 pumps. With these measures in place, the NRC staff determined that defense against human errors will be reasonably achieved.

7. The intent of plant design is maintained

   The SX system was designed to meet the requirements of general design criteria 44. Two trains of SX meet the requirements. With the SX A pump and train inoperable, the SX B train can mitigate all design basis accidents and events. A LOCA is not postulated because of the significant improbability of such an event during the 200-hour CT. But, if the SX train B failed during a LOOP initiating event, the licensee has sufficient redundancy by cross-connecting SX from Unit 1 to mitigate the event. The other design criteria for the SX system, including American Society of Mechanical Engineers (ASME) requirements, are not being challenged by the LAR. Therefore, the NRC staff considers intent of plant design to be maintained for this one- time 200-hour extended outage.

Based on the above, the NRC staff finds that the guidance in RG 1.177 has been met.

3.1.3 Key Principal 3: Safety margins

The extended CT is not in conflict with Codes and Standards approved for use by the NRC relevant to the SX system and associated supported systems. Safety analysis acceptance criteria as specified in the UFSAR, particularly, for large break LOCAs, are met during the extended CT, assuming the SX B train does not fail. The probability of a LOCA and failure of the SX B train during the 200-hour outage is highly unlikely and within the core damage frequency criteria approved by the NRC. Therefore, the NRC staff considers the safety margins are maintained.

Based on the above, the NRC staff finds that the guidance of SRP, Chapter 16.1, has been met.

With the 2A SX pump inoperable for repair, the plant can mitigate all DBAs, including a LOCA, assuming no failure in the 2B SX train. The probability of a LOCA and single-failure in the 2B SX train in the 200-hour CT is significantly small and less than NRC CDF criteria thus will not be considered. In the event of a LOOP and failure of the 2B SX train, the NRC staff has evaluated the proposed CT extension, using the traditional engineering considerations of RG 1.177. With no equipment failure and a LOOP occurring during the 200-hour CT, decay heat would be removed with the diesel-driven or a motor-driven AFW pump and SG power-operated relief valves and the 2B SX train would be available to provide plant cooldown capability. If the 2B SX train also failed with the LOOP, the plant has provided compensatory measures using SX cross-connect from Unit 1 to provide enough SX flow to satisfy all necessary SX loads on Unit 2 and a full train of SX loads for Unit 1. With the SX supplied by the cross-connect from Unit 1 to Unit 2, Unit 2 will be able to operate a Unit 2 EDG and all loads cooled by SX in order to control primary

plant pressure and inventory, AFW and SG pressure control for decay heat removal, residual heat removal (RHR) shutdown cooling if necessary, RCP seal injection and seal cooling, and containment temperature control. With the above considerations, the staff concludes that the traditional engineering considerations of RG 1.177 to include: (1) balance of prevention of core damage and containment failure, (2) not over relying on programmatic activities, (3) having system redundancy, independence and diversity commensurate with the risk, (4) not having the potential of common cause failure, (5) maintaining physical barriers, (6) maintaining defense against human errors, (7) maintaining the intent of plant design, (8) not reducing safety margin and (9) being in compliance with current regulations, have been satisfied. Therefore, the NRC staff concludes that the licensee' LAR meets the regulatory requirements and guidelines specified in Paragraph 2.3, Regulatory Requirements and is thus acceptable.

3.1.4 Key Principle 4: Change in risk consistent with the Commission's Safety Goal Policy Statement

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making: that when the proposed changes result in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The NRC staff evaluation of Key Principal 4 for the proposed one-time TS change is described below.

Tier 1: PRA capability and insights

The first-tier evaluates the impact of the proposed change on plant operational risk. The Tier 1 review involves two aspects: (1) evaluation of the technical adequacy of the Braidwood PRA model and its application to the proposed change, and (2) evaluation of the PRA results and insights based on the licensee's proposed change.

PRA technical adequacy

RG 1.174, Revision 2, states that, "[t]he scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process." The technical adequacy of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the technical adequacy of the PRA. This applies to Tier 1, and it also applies to Ties 2 and 3 to the extent that a PRA model is used.

RG 1.200, Revision 2, describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for light-water reactors. RG 1.200, Revision 2, clarifies the ASME/American Nuclear Society (ASME/ANS) PRA standard to be ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,". The ASME/ANS PRA standard provides technical supporting requirements in terms of three Capability Categories (CCs). The intent of the delineation of the CCs within the supporting requirements (SRs) is generally that the degree of

scope and level of detail, the degree of plant specificity, and the degree of realism increase from CC I to CC III. In general, the NRC staff anticipates that current good practice, i.e., CC II of the ASME/ANS standard, is the level of detail that is adequate for the majority of applications.

Internal events of PRA (including internal flooding)

The Braidwood 2015 PRA model (BB011b4) is a full-power, internal events PRA model that addresses both core damage frequency (CDF) and large early release frequency (LERF) and was used to evaluate the impact of the proposed change on plant risk for internal events and internal floods. The licensee's risk management process for maintaining and updating the PRA ensures that the PRA model remains an accurate reflection of the as-built and as-operated plant. PRA model BB011b4 is based on incorporating the RCP safe shutdown seals installed at Braidwood into internal events PRA model BB011b, which was peer reviewed in 2013.

Internal events PRA model BB011b has been subjected to a full-scope peer review in July 2013 in accordance with ASME/ANS RA-Sa-2009 , and RG 1.200, Revision 2. Details about this peer review, including peer-review findings and their disposition, are discussed in the September 30, 2016 submittal, as supplemented by the licensee's response to an RAI dated October 28, 2016, to NRC staff follow-up questions. Table 4-2 in Attachment 5 to the licensee's September 30, 2016, letter, identified 6 SRs that were evaluated as not meeting CC II and 10 SRs that were assessed as being at CC I. Table 4-3 in Attachment 5 to the LAR lists 24 peer review findings that are associated with SRs assigned to be at least CC II from the peer review consistent with the
RG 1.200, Revision 2.

Section 3.2.3 of Attachment 5 to the LAR describes a sensitivity analysis performed by the licensee to address peer review finding IFSO-A4-01 related to the absence of supporting information for not considering maintenance-induced flooding. In this analysis, the licensee increased all internal flooding frequencies by a factor of 1.45 to re-calculate the ICCDP and ICLERP for the 2A SX extended CT. The sensitivity analysis results show a minimal change in ICCDP and ICLERP. Therefore, it is determined that this open facts and observation (F&O) does not have a significant impact on the evaluation for extending the SX 2A CT.

Based on the NRC staff evaluation of the licensee's disposition of the F&Os in Section 4.0 of Attachment 5 to the LAR, the sensitivity analysis of finding IFSO-A4-01 in Section 3.2.3, and the licensee's response to RAI 17 in its letter dated October 28, 2016 the NRC staff concludes that all F&O findings associated with the Braidwood internal events PRA model were properly assessed and dispositioned to support the internal events PRA technical adequacy for the proposed one-time CT extension. Also, the large margin in the results of this risk assessment (i.e., ICCDP and ICLERP for the proposed one-time CT extension are much less than the acceptance criteria in RG 1.177) provides high confidence that any uncertainty associated with the Braidwood internal events PRA model would not change the conclusions of this assessment.

Fire PRA

The Braidwood Fire PRA (FPRA) model BB011b-FL-B, used for the risk evaluation of the SX 2A pump CT extension, is an interim implementation of NUREG/CR-6850 (ADAMS Accession No.

ML052580118) and other approved methodologies. The FPRA model was peer-reviewed in May 2015 by a review team assembled by the Pressurized-Water Reactor Owners Group (PWROG) in accordance with RG 1.200, Revision 2. Based on the NRC staff evaluation of the licensee's disposition of the F&Os associated with this peer review, the staff concludes that all F&O findings associated with the FPRA were properly assessed and dispositioned to support the FPRA technical adequacy for the proposed one-time CT extension. The FPRA model includes a full scope representation from the risk of fire for all Braidwood site fire areas. The selection of the global plant analysis boundary and the criteria for including/excluding plant areas are consistent with the current NUREG/CR-6850 guidance and methods. Therefore, the scope of areas included in the FPRA is sufficient for this application. Fire scenario development in the FPRA model includes fixed ignition sources and transient sources, consistent with the guidance in NUREG/CR-6850. The potential for main control room abandonment due to environmental conditions is also included in the model. Specific consideration of hot gas layer, multi-compartment analysis, and the potential for multiple spurious operations is also included in the FPRA. In addition, instrumentation has been explicitly included in the FPRA. In Section 3.5.3 of Attachment 5 to the LAR, the licensee discusses in detail the completeness of the FPRA model and concludes there were no major form of completeness uncertainty that would change the results of this assessment.

The licensee has identified the top fire zones that contribute 5% or more of the total fire risk:

- 5.1-2 Division 22 ESF Switchgear Room 26 percent
- 5.1-1 Division 12 ESF Switchgear Room 10 percent
- 3.2-0 Auxiliary Building El. 439'-0" 9 percent
- 11.4-0 Auxiliary Building General Area, El. 383' 6 percent
- 11.6-2 Division 22 Containment Electrical Penetration Area, El. 426' 5 percent

Based on reviewing the cutsets and importance measures associated with core damage as a result of fire with 2A SX out of service (OOS), the top cutsets are associated with failure to trip the RCPs or the loss of RCP seal cooling. Section 3.3.2 of Attachment 5 to the LAR states that "The significant operator actions are related to RCP trip/seal cooling, SX unit cross-tie, refueling the diesel-driven AFW day tank, and manual operation of AFW control valves. These operator actions confirm the impact of fire on SX potentially resulting in failure of RCP thermal barrier cooling and cooling for seal injection pumps. Operator briefings on the importance of these actions will be performed prior to entering the 2A SX OOS configuration."

The modeling of the RCP shutdown seals follows the guidance in PWROG-14001-P, "PRA Model for the Generation III Westinghouse Shutdown Seal," Revision 1 dated July 3, 2016 (ADAMS Accession No. ML14190A332). Since a loss of SX may impact RCP seal cooling and challenge the RCP seals, the modeling of the shutdown seals and associated human actions is identified as a potentially key uncertainty for this application. In the response to RAI 15 in its letter dated October 28, 2016), the licensee confirmed that the new Westinghouse Generation III seals have been installed in all RCPs at Braidwood. The licensee also performed a

sensitivity analysis to double the failure probability of the Generation III seals due to the uncertainty of their failure rate. The results from the sensitivity analysis showed that the fire ICCDP (2.7×10^{-6}) and the total ICCDP (2.8×10^{-6}) are significantly below the acceptance guidelines of RG 1.177, Revision 1. In the response to RAI 15, the licensee also clarified that there is sufficient time for operators to trip the RCP pumps to prevent damage to the shutdown seals or to protect the non-shutdown seals. In addition, the response to RAI 15 clarified that the detailed human reliability analysis was performed using guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," November 30, 2009.(ADAMS Accession No. ML093350494) to evaluate human error probabilities for a range of fire scenarios, including that of tripping the RCP pumps.

There are adequate compensatory actions in place to reduce the risk from fire events during the extended CT of SX 2A. These compensatory actions are discussed in Section 3.2.4.2 of this safety evaluation (SE). Based on NRC staff review of the technical adequacy of the Braidwood FPRA associated with the LAR and the licensee's response to RAI 15 in its letter dated October 28, 2016), the NRC staff concludes that the Braidwood FPRA is sufficiently adequate to provide risk insights for the proposed one-time CT extension. Also, the large margin in the results of this risk assessment (i.e., ICCDP and ICLERP for the proposed one-time CT extension are much less than the acceptance criteria in RG 1.177) provides high confidence that any uncertainty associated with the Braidwood FPRA model would not change the conclusions of this assessment.

Seismic and other external hazards

Braidwood does not have a quantitative seismic PRA model. Therefore, the licensee performed a qualitative assessment to evaluate seismic risk. Section 3.3.3 of Attachment 5 to the LAR presents the qualitative analysis to screen seismic risk as a non-significant contributor to the risk assessment of the proposed one-time CT extension. To support the qualitative assessment of seismic risk, the licensee used seismic risk insights, representative SSCs fragility information, and seismic walk-down information that were developed in 2013 for the Phase 1 seismic PRA. The licensee concluded that the additional risk due to a seismic event is qualitatively evaluated as low and would not have a significant impact on the overall results or conclusion for this risk evaluation.

The licensee has screened external floods, high winds and tornadoes, and other external hazards as non-significant contributors to the risk of the proposed CT extension based on the Braidwood Nuclear Generating Station, Units 1 & 2, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities Submittal Report," NRC Docket Nos 50-454 and 50-455, Common Wealth Edison Company, June, 1997 (ADAMS Accession No. ML080080366), and the guidance provided in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, Final Report," June 30, 1991 (ADAMS Accession No. ML063550238). The screening evaluations have been made based on the Braidwood Updated Final Safety Analysis Report and NRC staff's Final Safety Evaluation for Braidwood IPEEE ((ADAMS Accession No. ML011520003).

Regulatory Position 2.3.2 of RG 1.177 states that the scope of the analysis should include all hazard groups (i.e., internal events, internal flood, internal fires, seismic events, high winds,

transportation events, and other external hazards) unless it can be shown that the contribution from specific hazard groups does not affect the decision. The NRC staff finds that the licensee followed RG 1.177 by providing a qualitative analysis of seismic impact, performing screening evaluations of other external hazards and determining that those hazards do not impact this application. In addition, the compensatory actions listed in the LAR would reduce any risk associated with these external hazards.

Sensitivity and uncertainty analyses

Regulatory Position 2.3.5 of RG 1.177 states that the risk resulting from TS CT changes is relatively insensitive to uncertainties, because uncertainties associated with CT changes tend to similarly affect the base case and the change case.

Section 3.5 of Attachment 5 of the LAR present the uncertainty and sensitivity analyses associated with the risk evaluation for one-time CT extension. The LAR addresses parameter, modeling, and completeness uncertainties. The parameter uncertainty assessment indicates that the direct use of point estimate CDF/LERF results, as compared to the use of mean CDF/LERF results, is acceptable for this assessment. In the response to RAI 14 stated in the licensee's letter dated October 28, 2016, the licensee also clarified that truncation levels set for the internal events results listed in Table 3.2-1 and Table 3.2-9 of Attachment 5 to the LAR and the FPRA evaluation results listed in Section 3.3.2 of Attachment 5 to the LAR demonstrated a less than 5 percent change in risk from the previous decade in truncation level in accordance with the requirements stated in ASME/ANS RA-Sa-2009.

The large margin between the risk evaluation results and the ICCDP and ICLERP acceptance guidelines in RG 1.177, in addition to the multiple compensatory measures that the licensee committed to implement, provide confidence that modeling uncertainty would not change the conclusions of the risk evaluation. Based on the discussion above, the NRC staff finds that the licensee's assessment of sensitivity and uncertainty is consistent with RG 1.177.

PRA results and insights

The licensee evaluated the impact of the proposed change on plant risk for internal events, internal flooding, and internal fires using the Braidwood internal events PRA model (BB011b4) and the FPRA model BB011b-FL-B. The licensee has demonstrated that the one-time TS CT change has only a small quantitative impact on the plant risk.

This risk evaluation is specific to the Braidwood 2A SX pump outage with all relevant configurations represented in the PRA model assumptions stated in Section 3.1.2 of Attachment 5 to the LAR:

- The SX 2A Pump CT is assumed to increase from its current duration of 72 hours to a proposed duration of 200 hours.
- The base analysis in this risk assessment assumes one entry per year into the proposed CT for purposes of calculating changes in annual CDF. This is consistent with the current plans to enter the extended CT only once for a pump replacement/repair.

- This risk assessment does not credit the averted online risk due to a forced shutdown that would be required due to exceeding the existing CT.
- The PRA analysis assumes unavailability of the SX 2A pump via its corresponding maintenance basic event.
- No elective maintenance will be performed on the SX 1A, 1 B, or 2B pumps. These maintenance terms are set to FALSE for the quantification.
- There will be no elective maintenance work on the 1A, 1B, 2A, or 2B emergency diesel generators during the 2A SX extended CT. These maintenance terms are set to FALSE for the quantification.
- There will be no elective maintenance work on the Unit 2 AFW pumps. These maintenance terms are set to FALSE for the quantification.
- There will be no elective maintenance on the RCFC SX inlet valves, SX 16A/B, or the RCFC SX outlet valves, SX 27A/B, on either unit due to interlocks that could prevent use of the remaining SX pumps. These maintenance terms are set to FALSE for the quantification.
- There will be no elective maintenance on the 211, 212, 213, or 214 instrument busses or their associated inverters and transformers. The inverter and transformer maintenance terms are set to FALSE for the quantification.
- Additional elective maintenance activities will be prohibited during the repair as compensatory measures to reduce plant risk that are not included in the quantification results. The complete list of restricted maintenance, protected equipment, and additional compensatory measures is summarized in Section 5.4.1 of Attachment 5 to the LAR.

To determine ICCDP and ICLERP, the licensee calculated the CDF and LERF for the baseline case using the Braidwood internal events PRA model (BB011b4) and the FPRA model BB011b-FL-B based on average test/maintenance unavailability for all equipment. The CDF and LERF were calculated for the one-time CT extension case using the average maintenance models and setting the appropriate basic events in these models to represent the extended CT configuration, which are stated in Table 3.1-2 in Attachment 5 to the LAR.

The licensee reported the following results in Table 3.4-1 in Attachment 5 to the LAR by calculating the total ICCDP and ICLERP (for internal events, internal flooding, and fire events) that reflect the entire 200 hours CT for the 2A SX pump outage.

ICCDP= 2.7×10^{-6} (RG 1.177 Acceptance Guideline: $< 1 \times 10^{-5}$ with effective compensatory measures implemented to reduce the sources of increased risk)

ICLERP = 4.0×10^{-8} (RG 1.177 Acceptance Guideline: $< 1 \times 10^{-6}$ with effective compensatory measures implemented to reduce the sources of increased risk)

Based on the above, the NRC staff finds that the licensee meets appropriate risk measures specific to one-time only CT changes considering the compensatory measures discussed in Section 3.2.4.2 of this SE, and, therefore, meets the guidance of RG 1.177, RG 1.174, and RG 1.200.

Tier 2: Avoidance of Risk-Significant Plant Configurations

Under the Tier 2 acceptance guideline in RG 1.177, the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

Based on configuration-specific risk insights provided by the Braidwood internal events PRA model BB011b4, the FPRA model BB011b-FL-B, and as part of the Braidwood configuration risk management program (CRMP), the licensee identified risk-significant combinations of equipment that if out-of-service during the 2A SX pump outage would significantly increase risk, and identified further compensatory actions and restrictions for entry into preventative maintenance (PM) to avoid high risk equipment OOS combinations during that time. Section 5.4.1 of Attachment 5 to the LAR, as clarified in the licensee's letter dated October 28, 2016, in its response to RAI 16, the licensee discusses in detail these compensatory measures that will be implemented during the planned SX configuration to assure the risk impacts are acceptably low. These compensatory measures include:

1. There will be no elective maintenance work on the remaining SX pumps, (1A, 1 B, 2B) during the 2A SX extended CT. Additionally, this equipment will be protected for this one-time outage. This supports the maintenance assumptions in the risk analysis.

2. There will be no elective maintenance work on the EDGs (1A, 1B, 2A, 2B) during the 2A SX extended CT. Additionally, this equipment will be protected for this one-time outage. This supports the maintenance assumptions in the risk analysis and also supports mitigation of a loss of offsite power (LOP) during the maintenance window.

3. There will be no elective maintenance work on the Unit 2 AFW pumps (2A, 2B). This equipment will be protected for the one-time outage. This supports the maintenance assumptions in the risk analysis.

4. There will be no elective maintenance on the 1/2SX16A/B (i.e., RCFC SX inlet valves) and 1/2SX27A/B (RCFC SX outlet valves) due to interlocks that could prevent use of the remaining SX pumps. This supports the maintenance assumptions in the risk analysis.

5. There will be no elective maintenance on the 211, 212, 213, or 214 instrument busses or their associated inverters and transformers. Additionally, this equipment will be protected for the one-time outage. This supports the maintenance assumptions in the risk analysis.

6. There will be no elective maintenance on the startup feedwater pump, 2FW02P.

7. There will be no elective maintenance activities on the Unit 2 station auxiliary transformers.

8. The extended weather forecast' will be examined to ensure severe weather conditions are not predicted prior to entry into the CT. In the event of an unforeseen severe weather condition due to rapidly changing conditions, such as severe high winds, a briefing with crew operators will be performed to reinforce operator actions and responses in the event of a LOP.

9. Fire Risk Management Actions applicable for the SX 2A pump will be completed per OP-AA-201-012-1001 "OPERATIONS ON-LINE FIRE RISK MANAGEMENT" (these actions protect against fire impacting key redundant equipment).

10. Operations will hold briefings on the following actions:

    - On a loss of all RCP seal cooling, Operations trips RCPs in time to prevent damage to the Shutdown Seals relied on for extended loss of seal cooling events.

    - On a post-trip loss of AFW, Operations initiates flow from either the motor-driven feedwater pump (2FW01PA) or the startup feedwater pump (2FW02P) to at least one SG prior to reaching dry SG conditions.

    - Operators manually throttle 0/2SX007 valves when the RHR heat exchangers are used for emergency core cooling recirculation.

    - On a loss of Unit 2 SX, Operations opens the 1/2SX005 valve(s) to crosstie SX between the units.

    - Operations refills the diesel-driven AFW day tank from the fuel oil storage tank in order to maintain operation of the diesel-driven AFW pump.

    - On loss of vital instrument bus (120 VAC) 211 or 214, Operations opens the AFW flow control valves 2AF005A-D ("A" train) or 2AF005E-H ("B" train) by locally failing air to the valve operators, then Operations throttles 2AF013A-D ("A" train) or 2AF013E-H ("B" train) from the main control room to control SG levels.

11. Prior to entering the TS 3.7.8 Action Statement for repair of the 2A SX pump, an operating crew shift briefing and pre-job walk-downs will be conducted during the extended SX 2A outage window to reduce and manage transient combustibles and to alert the NRC staff about the increased sensitivity to fires in the fire zones listed in Section 5.4.1 of Attachment 5 to the LAR. Operating crew shift briefings will continue to be conducted every shift throughout the duration of the CT period. Additionally, planned hot work activities in these fire zones will be prohibited during the time within the extended SX 2A CT. In the event of an emergent issue requiring hot work in one of the listed zones, additional compensatory actions will be developed to minimize the risk of

fire. The listed fire zones were identified based on risk significance in the FPRA results (generally zones with Division 2 equipment that impact SX). (The purpose of these walk-downs is to reduce the likelihood of fires in these zones by limiting transient combustibles, ensuring transients, if required to be present, are located away from fixed ignition sources and eliminating or isolating potential transient ignition sources, e.g., energized temporary equipment and associated cables.)

Based on the above, the NRC staff finds that the licensee provided adequate analyses of risk-significant configurations while 2A SX pump is OOS and identified appropriate compensatory actions that can mitigate corresponding increases in risk. Additionally, should an emergent condition arise such that plant equipment in addition to the planned OOS equipment becomes unavailable, the associated risk will be assessed and managed in accordance with the Tier 3 program. Therefore, the staff concludes that the licensee's analysis of risk significant combinations and identification of compensatory actions meet the guidance of RG 1.177 and provide reasonable assurance that risk-significant plant equipment outage configurations will not occur during the PM.

Tier 3: Risk-informed configuration risk management

RG 1.177 states that Tier 3 is the establishment of an overall CRMP to ensure that other potentially lower probability, but nonetheless risk-significant, configurations resulting from maintenance and other operational activities are identified and managed. RG 1.177 further states that the licensee program for compliance with 10 CFR 50.65(a)(4) ensures that the risk impact of out-of-service equipment is appropriately assessed and managed.

The licensee states in Attachment 1 to the LAR that Braidwood has an established CRMP that implements 10 CFR 50.65(a)(4) requirements. The Braidwood CRMP requires "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. Braidwood currently has 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 re-assessed if an equipment failure/malfunction or emergent condition produces a plant configuration that has not been previously assessed."

Based on the above, the NRC staff finds the licensee's Tier 3 program for complying with 10 CFR 50.65(a)(4) is consistent with the guidance of Section 16.1 of the SRP and RG 1.177 and, thus, is acceptable.

<u>Conclusions</u>

The licensee has demonstrated that the technical adequacy and scope of its PRA models can support the LAR for the proposed one-time CT change to TS Section 3.7.8, Required Action A.1. The risk metrics used to support the LAR are consistent with RG 1.177. The NRC staff finds that the licensee has followed the three-tiered approach outlined in RG 1.177 to evaluate the risk associated with the proposed one-time TS CT change and, therefore, the proposed one-time TS change satisfies the fourth key safety principle of RG 1.177.

3.1.5 Key Principle 5: Performance measurement strategies – implementation and monitoring program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the changes to the TS. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change.

RG 1.177 states that the licensee is to use a three-tiered approach in implementing the proposed TS CT change. Application of the three-tiered approach is in keeping with the fundamental principle that the proposed change is consistent with the DID philosophy. Application of the three-tiered approach provides assurance that DID will not be significantly impacted by the proposed change. Furthermore, RG 1.177 states that, to ensure that extension of a TS CT does not degrade operational safety over time, the licensee should ensure, as part of its Maintenance Rule program (10 CFR 50.65), that when equipment does not meet its performance criteria, the evaluation required under the Maintenance Rule includes prior related TS changes in its scope.

The licensee provided a brief evaluation of the proposed TS change against the three-tiered approach in Attachment 1 of the LAR. In addition, in Attachment 1 to the LAR, the licensee confirmed that the SX pumps are monitored under the Braidwood Maintenance Rule Program. The licensee states in Attachment 1 to the LAR that "If the pre-established reliability or availability performance criteria are exceeded for the SX pumps, they are considered for 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1) actions, requiring increased management attention and goal setting in order to restore their performance (i.e., reliability and availability) to an acceptable level. The performance criteria are risk-informed and, therefore, are a means to manage the overall risk profile of the plant. An accumulation of large core damage probabilities over time is precluded by the performance category (i.e., the SX pumps are meeting established performance goals). Repair of the 2A SX pump is not anticipated to result in exceeding the current established Maintenance Rule criteria for the SX pumps."

Based on the above, the NRC staff finds that the implementation and monitoring program for the proposed TS change described by the licensee meets the guidance of the fifth key safety principle of RG 1.177 and meets the guidance in RG 1.174, RG 1.177, and SRP, Sections 19.1 and 16.1.

3.2 <u>Description of Operator Actions and their Risk Significance</u>

Risk Assessment

The NRC staff performed a quantitative assessment of the risk significance. Specifically, the risk significance was assessed of any human actions (HA) that the licensee may have to rely upon to crosstie to the opposite unit essential service water (SX) system. The NRC staff assessed (a) the plant baseline core damage frequency (CDF) and large early release frequency (LERF), (b) the frequencies of various initiating events requiring the SX system to be cross-tied to the opposite unit, and (c) the consequences of an operator failing to perform this

manual action. Further, the risk importance measures associated with this HA given the requested plant configuration and the expected duration was assessed.

NUREG-1764, "Guidance for the Review of Changes to Human Actions" provides guidance for performing risk assessments when changes to the plant create, modify, or affect task demands. The requested change modifies the task demands by making the task more likely to occur and places greater emphasis on correct performance on the human action.

This HA is currently modeled in the plant specific Braidwood SPAR model. Quantification of risk importance of the HA was performed using the current version of the Braidwood SPAR model.

The licensee submitted an incremental conditional core damage probability (ICCDP) value of 2.7E-6 which was consistent with the SPAR model results. In this case, a failure of the associated HA would result in the loss of a system (SX) supporting a key safety function (Decay Heat Removal) and subsequently would result in total failure of the key safety function. However, the HA failure would be self-revealing or the error would have to be repeated multiple times for any substantial consequences to result.

Based on the ICCDP, importance measures of the HA, and qualitative information provided in the licensee's September 30, 2016, submittal, the above assessment supports the conclusion that a Level II review is appropriate for this license amendment, the second highest of the graded reviews possible under the guidance in NUREG-1764.[1] The NRC staff requested additional information (E-mail from Joel Wiebe, dated October 20, 2016 (ADAMS Accession No. ML16294A343)) consistent with a Level II review.

Based on the licensee's October 26, 2016, response to the request for additional information, the NRC staff determined that there are no changes to operator manual actions as a result of this amendment. The operator actions used to cross-tie the units are the same as those that operators currently train for using existing procedures. The licensee indicates that all licensed operators at the unit have recently been trained on the operator actions needed to cross-tie the units between June 14, 2016, and July 15, 2016. In addition, the licensee described an operating experience review, indicated the results, and describes reasonable compensatory actions to prevent/mitigate any likely errors.

The NRC staff determined there are no changes to human-systems interfaces as a result of this amendment, nor are there changes to operator manual actions. The operator manual actions are consistent with the existing licensing basis and since they are unchanged as a result of this amendment, require no additional review.

The NRC Staff found that the operator manual actions described are identical to those described in the current licensing basis. Since there are no changes to the actions or to factors that may influence operator success (such as procedures, timing of actions, training, HSIs, etc.) it is reasonable to assume that the assumptions of the existing licensing basis remain

1 Note: The APHB assessment of risk is only for purposes of scoping the APHB review and may differ from the licensee's assessment of risk importance, and should not be considered as an accurate assessment of risk for other purposes, especially, those using plant-specific data and NRC-accepted methods of Probabilistic Risk Analysis and Human Reliability Analysis, PRA/HRA.

acceptable and the guidance of SRP Chapter 18, "Human Factors Engineering," Revision 2, NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, and NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, continue to be met.

3.3 Technical Specification Change

Existing TS 3.7.8 ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 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

Revised TS 3.7.8 ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One unit-specific SX train inoperable.	-------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. A.1 --------NOTE-------- Not applicable to Unit 2 during repair of the 2A SX pump during the one-time Unit 2 planned SX System outage to be completed no later than January 23, 2017. -------------------- Restore unit-specific SX train to OPERABLE status.	 72 hours
	OR	
	A.2 --------NOTE-------- Applicable to Unit 2 during repair of the 2A SX pump during the one-time planned SX System outage to be completed no later than January 23, 2017. Allowance of the extended completion time is contingent on meeting the compensatory measures described in EGC submittal letter RS-16-197. -------------------- Restore unit-specific SX train to OPERABLE status.	 200 hours

The revised TS allows a one-time planned SX system outage of 200 hours to be completed no later than January 23, 2017. The NRC staff questioned in its October 20, 2016 e-mail whether the TS was intended to allow multiple entries into the ACTION statement or would one entry be made during the proposed timeframe. In its October 26, 2016, supplement, the licensee stated:

> "Because the proposed CT extension is a one-time change, there will be no repeated entry into the Required Action associated with the proposed CT extension. When the 2A SX train is declared inoperable in support of the 2A SX pump repair described in Reference 1 (RS-16-197) [the licensee's letter dated September 30, 2016] it will remain inoperable until the pump is repaired, tested and the 2A SX train is declared operable.

Once the 2A SX train is declared operable, the proposed Note for Required Action A.1 and Required Action A.2 will no longer be applicable. This one-time entry is stated in the TS Notes. . ."

The NRC staff finds that while the revised TS extends the completion time for the one-time maintenance outage of the 2A SX system, it is acceptable based on the NRC findings in Sections 3.1 and 3.2, above. Therefore, the NRC staff finds that the revised TS continue to meet the requirements of 10 CFR 50.36(c)(2).

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC's regulations in 10 CFR 50.92 state that the NRC may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes have been evaluated using the risk informed processes described in RG [Regulatory Guide] 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 dated May 2011, RG 1. 177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1 dated May 2011 and NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 dated March 2009. The risk associated with the proposed change was found to be acceptable.

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The SX System is not considered an initiator for any of these previously analyzed events. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not require any physical change to any plant SSCs nor does it require any change in systems or plant operations. The proposed one-time increase in the CT [completion time] is consistent with the philosophy of the current TS LCO [limiting condition for operation] which allows one SX train to be inoperable for 72 hours. This change only extends the 72 hour CT to 200 hours which has been shown to be acceptable from a risk perspective. The minimum equipment required to mitigate the consequences of an accident and/or safely shut down the plant will be Operable or available during the extended CT. The proposed change is consistent with the safety analysis assumptions and resultant consequences. Based on the above, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change does not alter any existing setpoints at which protective actions are initiated and no new setpoints or protective actions are introduced. The design and operation of the SX System remains unchanged. The risk associated with the proposed increase in the time the 2A SX pump is allowed to be inoperable was evaluated

> using the risk informed processes described in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 dated May 2011, RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1 dated May 2011 and NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 dated March 2009. The risk was shown to be acceptable. Based on this evaluation, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, determined that the three standards of 10 CFR 50.92(c) are satisfied.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment on November 7, 2016. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility located within the restricted area as defined in 10 CFR Part 20. The Commission has previously issued a proposed finding (81 FR 72838; October 21, 2016) that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. The Commission has made a final determination that the amendment involves no significant hazards consideration. The NRC staff has determined that the amendment involves 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 not significant increase in individual or cumulative occupational radiation exposure. Accordingly, the amendment meets 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 amendment.

7.0 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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BGreen

Date of issuance: November 23, 2016

November 23, 2016

Mr. Bryan C. Hanson
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Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNIT 2 - ISSUANCE OF AMENDMENTS REGARDING 2A ESSENTIAL SERVICE WATER PUMP TECHNICAL SPECIFICATIONS FOR PUMP REPAIR (CAC NO. MF8438)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 191 to Renewed Facility Operating License No. NPF-77 for the Braidwood Station, Unit No. 2. The amendment is in response to your application dated September 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A474), as supplemented by letters dated October 26 and 28, 2016, and November 14, 2016 (ADAMS Accession Nos. ML16301A073, ML16302A468, and ML16319A397).

The amendment adds a Required Action A.2 that increases the completion time currently specified in Required Action A.1, "Restore unit-specific SX train to OPERABLE status," associated with Technical Specifications (TS) Section 3.7.8, "Essential Service Water (SX) System," from 72 hours to 200 hours. This proposed change will only be used one time during a planned 2A SX pump repair

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/
Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. STN 50-457

Enclosures:
1. Amendment No. 191 to NPF-77
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
PUBLIC LPL3-2 R/F
RidsRgn3MailCenter Resource
RidsNrrPMBraidwood Resource
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RidsNrrDorlLpl3-2 Resource
RidsNrrDorlDpr Resource
RidsAcrs_MailCTR Resource

ADAMS Accession NO. AMD: ML16315A302, BWI: ML16315A305 *via Memo

OFFICE	LPL3-2 /PM	LPL3-2/LA	SBPB/BC	STSB/BC	SRXB/BC	APLA/BC	APHB/BC	OGC	LPL3-2/BC
NAME	JWiebe	SRohrer	RDennig*	AKlein*	EOsterlie	SRosenberg*	SWeerakkody*	VHoang (NLO)	EMiller
DATE	11/16/16	11/14/16	11/7/16	11/15/16	11/18/16	11/9/16	11/10/16	11/22/16	11/23/16

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