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RS-16-222

October 28, 2016

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 2
Renewed Facility Operating License No. NPF-77
NRC Docket No. STN 50-457

Subject: Response to Request for Additional Information Related to License Amendment Request for a One-Time Extension of the Essential Service Water (SX) Train Completion Time to Support 2A SX Pump Repair

References:

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request for a One-Time Extension of the Essential Service Water (SX) Train Completion Time to Support 2A SX Pump Repair," dated September 30, 2016 (ML16274A474)

2) Email from J. Wiebe (NRC) to J. Bauer (Exelon Generation Company, LLC), "Initial RAIs Related to the Braidwood Unit 2 SX Pump Allowed Outage Time Amendment Request," dated October 20, 2016

3) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information License Amendment Request for a One-Time Extension of the Essential Service Water (SX) Train Completion Time to Support 2A SX Pump Repair," dated October 26, 2016

4) Email from J. Wiebe (NRC) to J. Bauer (Exelon Generation Company, LLC), "Supplemental Initial RAIs Related to the Braidwood Unit 2 SX Pump Allowed Outage Time Amendment Request," dated October 22, 2016

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Technical Specification (TS) 3.7.8, "Essential Service Water (SX) System," of Renewed Facility Operating License No. NPF-77 for Braidwood Station Unit 2. The proposed amendment would modify 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 train to OPERABLE status," from 72 hours to 200 hours. This proposed change is a one-time change to support a planned

2A SX pump repair scheduled to be performed before January 23, 2017. In Reference 2, the U.S. Nuclear Regulatory Commission (NRC) requested additional information related to its review of Reference 1. EGC responded to the Reference 2 request in Reference 3. In Reference 4, the NRC requested supplemental additional information related to its review of Reference 1. Attachment 1 to this letter provides the requested supplemental information. Attachment 2 includes a revised PRA analysis (previously included as Attachment 5 of Reference 1) in response to the NRC request.

EGC has also reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Reference 1, Attachment 1. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this letter and its attachments are being provided to the designated State of Illinois official.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact J. A. Bauer at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 28th day of October 2016.

Respectfully,

Patrick R. Simpson
Manager – Licensing
Exelon Generation Company, LLC

Attachments: 1. Responses to Request for Additional Information
2. BW-LAR-008, "Risk Assessment Input for the Braidwood One-Time Technical Specification Change for the Essential Service Water Pump 2A Completion Time from 72 to 200 Hours," Revision 1

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Responses to Request for Additional Information

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Technical Specification (TS) 3.7.8, "Essential Service Water (SX) System," of Renewed Facility Operating License No. NPF-77 for Braidwood Station Unit 2. The proposed amendment would modify 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 train to OPERABLE status," from 72 hours to 200 hours. This proposed change is a one-time change to support a planned 2A SX pump repair scheduled to be performed prior to January 23, 2017. In Reference 2, the U.S. Nuclear Regulatory Commission (NRC) requested additional information related to its review of Reference 1. EGC responded to the Reference 2 request in Reference 3. In Reference 4, the NRC requested supplemental additional information related to its review of Reference 1. The requested supplemental information is provided below. As noted in Reference 4, the numbering sequence continues from the number sequence from Reference 2.

<u>NRC RAI 12</u>

Regulatory Issue Summary 2007-06 states that the NRC staff expects that licensees fully address all scope elements with Regulatory Guide (RG) 1.200, Revision 2 by the end of its implementation period (i.e., April, 2010, one year after the issuance of RG 1.200, Revision 2). In accordance with RG 1.200, Revision 2, it is expected that the differences between ASME/ANS RA-Sa-2009 and earlier versions of the standard used in the internal events PRA peer review be identified and addressed.

Page 2 of the LAR states that the proposed change to the CT has been evaluated using the risk-informed processes described in RG 1.174, Revision 2 and RG 1.177, Revision 1. The risk associated with the proposed changes was determined to be acceptable. RG 1.177, Revision 1, Section 2.3.1, "Technical Adequacy of the PRA," states that "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." This section refers to RG 1.200, Revision 2, which endorses, with exceptions and clarifications, ASME/ANS RA-Sa-2009 as one acceptable approach for determining the technical adequacy of the PRA.

Section 1.1 of Attachment 5 to the LAR states that the analysis follows the guidance provided in RG 1.200, Revision 2. However, Section 4.4 of Attachment 5 to the LAR states that "[t]he ASME/ANS PRA Standard provides the basis for assessing the adequacy of the Braidwood PRA as endorsed by the NRC in RG 1.200, Revision 1. The predecessor to the ASME/ANS PRA Standard was NEI 00-02 which identified the critical internal events PRA elements and their attributes necessary for a quality PRA." Section 4.5 of Attachment 5 to the LAR states that a formal industry peer review was performed in July 2013 against Addendum B of the ASME/ANS PRA Standard, while, Section 4.6.2 of Attachment 5 to the LAR states that this formal peer review was performed against Addendum A of the ASME/ANS PRA Standard.

It is not clear whether RG 1.200, Revision 2, was used as the basis for assessing the technical adequacy of the Braidwood PRA used for this application. For the formal peer review of the internal events PRA in July 2013, clarify whether this peer review was performed against ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. Also, indicate whether this formal review was a full-scope or focused-scope review. If the peer review was performed against an earlier version of this guidance, then provide a gap assessment of the internal events PRA against ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. [Section 3.3, "Gap Assessment for PRAs Reviewed Against RG 1.200, Revision 1," of NEI 05-04, Revision 3, provides guidance on performing a gap assessment.]

EGC Response to RAI-12:

The reference to RG 1.200 Revision 1 (in Section 4.4 of Attachment 5 of Reference 1) is a typographical error. Additionally, reference to Addendum B (in Section 4.5 of Attachment 5 of Reference 1) should refer to Addendum A. These typographical errors have been corrected in the PRA application document (Attachment 5 of Reference 1) which is re-submitted as Attachment 2 of this response.

The Braidwood PRA peer review was performed to RG 1.200, Revision 2. The peer review report references ASME/ANS RA-Sa-2009. The peer review was a full-scope review.

NRC RAI 13

The LAR states that the proposed change to the Technical Specification completion time has been evaluated using the risk-informed processes described in RG 1.174, Revision 2, and RG 1.177, Revision 1. Attachment 5 to the LAR provides the supporting risk-informed evaluation of the requested change including an evaluation of the technical adequacy of the PRA in accordance with RG 1.200, Revision 2.

a) *RG 1.177, Revision 1, states that the proposed change must maintain sufficient safety margins. Section "SX Pump Repair/Replacement History" of Attachment 1 to the LAR (Page 5 of 24) states that the 1B SX pump "has been operating with minimal margin recently. Consequently, the rotating element will be replaced during Refueling Outage A1R19 (in September/October 2016) to increase the margin." The licensee's risk evaluation for this LAR assumes that the failure probabilities of all the SX pumps, except the 2A pump, remain the same (i.e., they all have sufficient margin and are not degraded).*

   Confirm that the rotating element of the 1B SX pump has been replaced during Refueling Outage A1R19. Also confirm that the 1A, 1B and 2B SX pumps have not degraded based on the latest surveillance test results.

   EGC Response to RAI-13a:

   The rotating element, including the pump shaft, impeller, impeller rings and case rings (referred to as the rotating element in Reference 1), was replaced on the 1B SX pump during the Braidwood Station refueling outage in Fall 2016 (A1R19).

   Following the repairs to the 1B SX pump during A1R19, a Comprehensive Inservice Test (IST) was performed as a Post Maintenance Test (PMT) on October 11, 2016 demonstrating an improvement in pump performance. The vibration data collected during the PMT was in the acceptable range as was the differential pressure (dP).

   The 1A and 2B SX pumps both had Group A (i.e., quarterly) IST surveillances performed in September of 2016. As shown in the table below, both pumps satisfactorily passed the Group A (i.e., quarterly) IST with vibration data and dP in the Acceptable Range. Thus, as shown by the results of the latest pump surveillances, the 1A, 1B, and 2B SX pumps are currently operating acceptably and are not degraded.

Pump	Most Recent Surveillance	dP	Acceptable Range	Within Range
1A	9/1/2016	74.2	70.2 < dP < 76.9	Yes
1B	10/11/2016	71.3	69.0 < dP < 73.4	Yes
2B	9/9/2016	72.9	71.7 < dP < 79.3	Yes

b) *Section 2.3.5, "Sensitivity and Uncertainty Analyses Relating to Assumptions in Technical Specification Change Evaluations," of RG 1.177, Revision 1, states that sensitivity analyses should be performed to address the uncertainties regarding important assumptions made in the submittal.*

If the 1A, 1B or 2B SX pump is in degraded condition, adjust the failure probability(ies) of the degraded pump(s) based on the existing margin and perform a sensitivity analysis to confirm that the conclusion of this LAR is not impacted.

EGC Response to RAI-13b:

Per the response to 13a, the 1A, 1B, and 2B SX pumps are not in degraded conditions. The pumps are all within operability limits and no adjustments to the failure probabilities are warranted.

NRC RAI 14

Section 2.3.3.4, "Truncation Limits," of RG 1.177 states that "truncation levels should be used appropriately to ensure that significant underestimation, caused by truncation of cutsets, does not occur as discussed below. Additional precautions relevant to the cutset manipulation method of analysis are needed to avoid truncation errors in calculating risk measures."

ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2, contains "High Level Requirements for Quantification," HLR-QU-B. Truncation limit is one of the HLR-QU-B quantification requirements. In Table 2-2.7-3(b) of ASME/ANS RA-Sa-2009, QU-B2 requires to "TRUNCATE accident sequences and associated system models at a sufficiently low cutoff value that dependencies associated with significant cutsets or accident sequences are not eliminated."; QUB3 requires to "ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no significant accident sequences are inadvertently eliminated. For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF or LERF, and the final change is less than 5%."

Explain how the truncations used for calculating the CDFs and LERFs listed in Table 3.2-1 and Table 3.2-9 in Attachment 5 to the LAR meet the above truncation limit requirements. If the truncations used in the LAR do not meet the ASME/ANS RA-Sa-2009 requirements, then update the risk evaluation of the LAR using appropriate truncation limits and explain how these new truncation limits meet the above truncation limit requirements.

EGC Response to RAI-14:

The Full Power Internal Events (FPIE) PRA model BB011b4 used in calculating the results in Table 3.2-1 and Table 3.2-9 was evaluated for convergence prior to model approval. The truncation levels (1E-10 for CDF and 1E-11 for LERF) demonstrated a less than 5% change from the previous decade in truncation level in accordance with PRA standard requirement HLR-QU-B (Reference 5 of Attachment 2).

The Fire PRA truncation sensitivity evaluation meets the criteria specified in Fire PRA standard requirement FQ-B1 (Reference 5 of Attachment 2, which applies the requirements of FPIE HLR-QU-B) which shows a less than 5% change in CDF and LERF from the previous decade in truncation level.

NRC RAI 15

Section 2.5.3 of RG 1.174, Revision 2, states that "The development of the PRA model is supported by the use of models for specific events or phenomena. In many cases, the industry's state of knowledge is incomplete, and there may be different opinions on how the models should be formulated. Examples include approaches to modeling human performance, common-cause failures, and reactor coolant pump [(RCP)] seal behavior upon loss of seal cooling. This gives rise to model uncertainty." Regarding the model uncertainty, Section 2.5.3 of RG 1.174, Revision 2, states that "the impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models." In addition, Section 2.5.5 states that "in general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (i.e., change generally remains in the appropriate region)."

Sections 3.2.3, "Peer Review Finding IFSO-A4-01 Sensitivity Analysis," and 3.5, "Uncertainty Assessment," of Attachment 5 to the LAR provide the uncertainty and sensitivity analyses related to the risk evaluation for this application. Section 3.5.2, "Model Uncertainty," of the LAR states that "[b]ecause a loss of SX can impact RCP seal cooling and lead to a challenge to the RCP seals, the modeling of the Shutdown Seals and associated human actions is identified as a potentially key uncertainty for this application." Although the LAR implies that the Generation III (GEN III) Westinghouse Shutdown Seals were used at Braidwood, there is no confirmation about the type(s) of RCP shutdown seals installed at Braidwood. It is unclear how many RCPs have been installed with the GEN III seals. The RCP shutdown seals are modeled in the PRA using the guidance in PWROG-14001-P, "PRA Model for the Generation III Westinghouse Shutdown Seal," Revision 1. Although PWROG-14001-P is the latest available industrial guidance for the Gen III seals, it has not been endorsed nor approved by the NRC. The likelihood and magnitude of inventory loss resulted from the Gen III seals failures may be greater than that assumed by this Westinghouse guidance. Given the issues identified above, provide additional information to the following:

a) *Describe the type(s) of RCP shutdown seals installed at Braidwood and whether the Gen III seals were installed on all the RCPs for Units 1 and 2.*

EGC Response to RAI-15a:

Braidwood Station has installed the Westinghouse Generation III SHIELD (also known as the RCP Passive Thermal Shutdown Seal (SDS)) on all the RCPs for Units 1 and 2.

Braidwood Station Unit 2 completed installation of the RCP seals in the refueling outage in Fall 2015 (A2R18). Braidwood Station Unit 1 completed installation during A1R19 in Fall 2016.

b) *Provide the results of a sensitivity analysis (i.e., updated values in LAR Tables 3.2-1 and 3.2-2 for internal events; LAR Tables 3.3-1 and 3.3-2 for fire; and LAR Table 3.4-1 for total risk and comparison to the acceptance guidelines) that considers reduced credit for the RCP shutdown seals (i.e., reduce credit by a factor of 2). Confirm that the results of this sensitivity analysis still meet the risk acceptance guidelines of RG 1.177, Revision 1.*

[Note: This sensitivity analysis should take into consideration any changes made to the PRA or risk evaluation as a result of addressing Items 2[13], 3[14] and 6[17] listed in this RAI.]

If RG 1.177 risk acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusion of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

<u>EGC Response to RAI-15b:</u>

A sensitivity study was performed for reduced credit for the RCP SDS. The failure probability for the seals was increased by a factor of 2. This increase did not lead to exceeding of the risk acceptance guidelines. The results are shown below:

Increased Failure Rate RCP Seal (2x) sensitivity			
FPIE ICCDP	7.7E-08	FPIE ICLERP	1.2E-09
Fire ICCDP	2.7E-06	Fire ICLERP	4.0E-08
Total ICCDP	2.8E-06	Total ICLERP=	4.1E-08

Items 2[13], 3[14], and 6[17] did not result in any analysis changes that required modification of this sensitivity study.

c) *In order to verify the adequacy of the human error probabilities (HEPs) associated with the human failure event (HFE) related to tripping the RCPs to preclude damage to the shutdown seals, which is a compensatory measure stated in the LAR, address the following:*

i. *Confirm that the HEPs for tripping the RCPs to preclude damage to the shutdown seals used in the internal events and fire PRAs reflect current plant procedures.*

<u>EGC Response to RAI-15c (i):</u>

The FPIE HEPs related to RCP trip on loss of cooling (1RC-PMTRIPAHSYOA, 1RC-PMTRIPBHSYOA) preclude damage by using system window timing based on the applicable PWROG Topical Report (PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," Risk Management Committee, PA-RMSC-

0499R2 July 2014) and manipulation timing based on operator interviews. The interviews were conducted at Byron Station, but the results are considered to be representative of Braidwood Station as the actions of the applicable procedure sections are consistent. The actions for tripping the RCPs use the current Braidwood Station procedures.

ii. For the fire PRA, explain how the HEPs were calculated for this HFE, and why they are reasonable for this application. Provide sufficient details (e.g., input assumptions and numerical values) to show how these HEPs were developed, including:

- *A discussion of the specific actions and instructions for tripping the RCPs to preclude damage to the shutdown seals for fire scenarios, including the cues or indications operators will use to initiate RCP trip. Provide a timeline for these operator actions, and how the time available and time required to complete operator actions were estimated.*

- *Confirm that the modeling and feasibility study of this HFE was performed consistent with guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," July 2012 (ADAMS Accession No. ML 12216A104). Otherwise, justify the basis for the human reliability analysis of this HFE.*

<u>EGC Response to RAI-15c (ii):</u>

Procedure 1(2)BWOA RCP-2 will guide the RCP tripping action response. The following annunciators will initiate the RCP tripping actions:

- RCP SEAL WTR INJ FLOW LOW (2-7-B2)
- RCP 2_ THERMAL BARR CC WTR FLOW LOW (2-7-_4)
- RCP LOWER BRNG TEMP HIGH (2-7-C2)
- RCP SEAL OUTLET TEMP HIGH (2-7-D3)

If these cues are not available because of fire, operators would monitor the high RCP thermal barrier Component Cooling (CC) water temperature (annunciator 2-7-E3). When high RCP thermal barrier temperature occurs, the procedure will direct the operators to manually trip the affected RCP(s). If all cues are lost, post fire shutdown procedures provide guidance to trip the affected RCP(s). Similar to the analysis performed to estimate the FPIE HEPs related to RCP trip on loss of cooling, system window timing is based on the applicable PWROG Topical Report (PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," Risk Management Committee, PA-RMSC-0499R2 July 2014). Manipulation timing and delay time due to fire is based on operator interviews performed at Braidwood Station in March 2016.

For the 1(2)RC-PMTRIPAHSYOA-F action (i.e., operators fail to trip RCP to prevent damage to the shutdown seals), a 13 minute time window is assumed in the Human Reliability Analysis (HRA) based on PWROG-14001-P Revision 1 and is the time available. Based on operator interviews, a manipulation time of 5 minutes is expected. An additional 5 minutes are added to account for any fire response

activity. The total expected time of 10 minutes is the time used in the HRA and is bounded by the PWROG-14001-P Revision 1 time of 13 minutes available.

For the 1(2)RC-PMTRIPBHSYOA-F action (i.e., operators fail to trip RCP to protect non-shutdown seals) a 30 minute time window is assumed in the HRA based on PWROG-14001-P Revision 1 and is the time available. Based on operator interviews, a manipulation time of 5 minutes is expected. An additional 5 minutes are added to account for any fire response activity. The total expected time of 10 minutes is the time used in the HRA and is bounded by the PWROG-14001-P Revision 1 time of 30 minutes.

Fire HEPs are based on FPIE HEPs with modifications per the NUREG-1921 methodology. Multiple and diverse cues are available for this function. Cues will trigger an RCP trip due to low injection/CC flow or high RCP seal outlet/lower bearing temperature. In addition, loss of all cues, due to fire, are expected to result in an RCP trip given guidance in Post Fire shutdown procedures to trip RCPs if valid flow/temperature indications are unavailable.

NRC RAI 16

Section 2.3.6 of RG 1.177, Revision 1, states that "if compensatory measures are considered as part of the analysis of the change, they should be included in the overall application for the TS change." Sub-section "Summary of Compensatory Measure Impacts on Important Fire Zones" (Pages 3-24 and 3-25) of Attachment 5 to the LAR listed three compensatory actions. However, Attachment 1, Sub-Section "Tier 2: Avoidance of Risk-Significant Plant Configurations" (Pages 11 and 12 of 24) of the LAR and Section 5.4.1, "Compensatory Measures," of Attachment 5 to the LAR does not include all of these measures. Also, Sub-section "Summary of Compensatory Measure Impacts on Important Fire Zones"(Pages 3-24 and 3-25) of Attachment 5 to the LAR refers to "BW-CRM-115, Exelon Risk Management Team, Development of Risk Management Actions for the Inclusion of Fire Insights into Braidwood Configuration Risk Management Program, BW-CRM-115, Revision 1, February, 2014"for configuration risk management, while Section 5.4.1 of Attachment 5 to the LAR refers to "OP-AA-201-012-1001, Operations On-Line Fire Risk Management, Revision 1"for fire risk management actions.

a) Describe the differences between the two references for risk management actions and clarify what compensatory measures will be implemented from these risk management actions during and prior to the repair of the 2A SX pump.

EGC Response to RAI-16a:

The compensatory measures as listed in Attachment 1 of Reference 1 are the compensatory measures that will be implemented prior to and during the repair of the 2A SX pump. The actions listed in Attachment 5 of Reference 1 are consistent in intent to the actions as written in Attachment 1 of Reference 1. During the final review of the compensatory actions listed in Attachment 1 of Reference 1, enhancements were made to the compensatory action text used in Attachment 5 of Reference 1. The purpose of these enhancements was to add specificity and detail to the compensatory actions with the intent of providing better direction to the operating crews. These enhancements did not impact the assumptions or results of the PRA analysis. The PRA analysis has been revised to reflect the enhanced text and is included as Attachment 2.

b) *Ensure that all the compensatory measures are consistent and that they are explicitly listed in the following sections of the LAR:*
   - *Attachment 1, Sub-Section "Tier 2: Avoidance of Risk-Significant Plant Configurations" of the LAR,*
   - *Attachment 5, Section 5.4.1, "Compensatory Measures," and*
   - *Attachment 5, Sub Section "Summary of Compensatory Measure Impacts on Important Fire Zones."*

EGC Response to RAI-16b:

The wording in Attachment 5 of Reference 1, Section 5.4.1, "Compensatory Measures" has been revised to match the list in Attachment 1, Sub-Section "Tier 2: Avoidance of Risk-Significant Plant Configurations" to ensure consistency (see Attachment 2 of this response). In addition, as discussed in Reference 3, response to NRC RAI-5, the last bulleted item of compensatory measure No. 10 referred to Vital Instrument Buses 111 and 114. The correct instrument buses are 211 and 214. This error has been corrected in the compensatory measures listed.

Additionally, the fire specific compensatory measures listed in Attachment 1 of Reference 1 were examined for consistency with Attachment 5 Reference 1, Sub Section "Summary of Compensatory Measure Impacts on Important Fire Zones," and the fire-specific Attachment 1 actions are contained in the Attachment 5 Sub-Section "Summary of Compensatory Measure Impacts on Important Fire Zones." The sub-section "Summary of Compensatory Measure Impacts on Important Fire Zones" included in Attachment 5 of Reference 1 (i.e., Attachment 2 of this response) has been changed and is now consistent with Attachment 1 of Reference 1. These clarifications do not impact the PRA analysis.

NRC RAI 17

Section 2.3.1, "Technical Adequacy of the PRA," of RG 1.177, Revision 1 states that the technical adequacy of the PRA must be compatible with the safety implications of the Technical Specification change being requested and the role that the PRA plays in justifying that change. RG 1.177, Revision 1 endorses the guidance provided in RG 1.200, Revision 2. RG 1.200, Revision 2 endorses, with exceptions and clarifications, ASME/ANS RA-Sa-2009 which provides technical supporting requirements in terms of three Capability Categories. The intent of the delineation of the Capability Categories within the Supporting Requirements is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from Capability Category I to Capability Category III. RG 1.200, Revision 2 describes a peer review process utilizing ASME/ANS RA-Sa-2009 as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision. The primary results of a peer review are the Findings and Observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os.

Table 4-3 of Attachment 5 to the LAR lists those "Findings" that are associated with supporting requirements (SRs) that were otherwise assigned to be at least Capability Category II from the peer review consistent with the RG-1.200, Revision 2. Address the following items related to the F&Os listed in Table 4-3:

a) *AS-B3-01 (Page 4-14): "Potential failure of containment sump suction screens due to debris clogging is not represented in the fault tree." The licensee states that no significant*

effect is expected, since dominant sequences often already have failure of recirculation for other reasons.

According to Table 2-2.2-3(b) in ASME/ANS RA-Sa-2009, SR AS-B3 requires "for each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models." This requirement is applicable to Capability Categories I to III.

Not including failure of containment sump suction screens due to debris clogging in the sequences produces non-conservative results. Explain why there is no significant effect on the modeling results used for this application.

EGC Response to RAI-17a:

In Table 4-3 of Attachment 5 of Reference 1, the assessment of impact of not representing sump clogging is that it "is not expected to be a significant effect since dominant sequences often already have failure of recirculation for other reasons." Loss of SX scenarios are key contributors, and these already fail the recirculation function due to lack of decay heat removal, so loss of the sump due to clogging is non-minimal. Loss of CC scenarios are similar. Large LOCAs are not significant contributors. Medium LOCAs and Small LOCAs (including induced LOCAs) are key contributors, but recirculation failure often occurs due to other causes more likely than sump clogging. Based on WCAP-16882-NP, Revision 1, sump clogging probability for Small LOCAs would be no more than 1E-5, and for Medium LOCAs, no more than 1E-4. As an example of failures that are more likely with the same impact, all LOCAs at Braidwood are assumed to require an operator action to adjust SX flow to the CC heat exchangers, which has a human error probability of 3.4E-3, greater than any sump clogging probability.

b) *DA-D7-01 (Page 4-15): "SR-DA-D7 requires that if common cause events deleted from common cause population of estimate formula due to non-applicability events in the total population also have to be screened and deleted if non-applicable. It was noted that one common cause event excluded from the common cause group was not excluded from the total population." The licensee states that "changes in the one CCF term identified by the review team do not have a large overall effect on model results."*

According to Table 2-2.6-5(d) in ASME/ANS RA-Sa-2009, DA-D7 requires "If screening of generic event data is performed for plant-specific estimation, ENSURE that screening is performed on both the CCF events and the independent failure events in the database used to generate the CCF parameters." This requirement is applicable to Capability Categories I to III.

Justify why the impact of not including that one common cause event is not significant for this application.

EGC Response to RAI-17b:

For DA-D7, the screened event from the plant-specific Common Cause Failure (CCF) calculation would impact the SX or Non-Essential Service Water (WS) strainers. WS CCF strainer events are insignificant contributors to the risk calculations. SX CCF strainer events are minor contributors, with the 4 of 4 CCF as the only significant event at ~2% FV. The alpha factor for 4 of 4 SX strainer CCF is calculated as 0.92/147 = 6.2E-3. If the one event is removed from the denominator, the calculation of the alpha factor becomes 0.92/146 = 6.3E-3, or a factor change of 147/146 = 1.007; i.e., less than 1% change in alpha factor. Combined with the low importance of strainer events, the impact on the risk calculations is negligible.

Table 4-4 of Attachment 5 to the LAR lists the "Findings" that are associated with Braidwood's fire PRA that did not meet Capability Category I Supporting Requirements. Address the following items related to the F&Os listed in Table 4-4:

c) *PRM-B15, F&O No.15-15 (Page 4-21): "BW-PRA-021.05, Fire PRA Plant Response Model notebook Rev 0, Section 3.1.8 and Appendix B. Appendix B documents the review of the containment paths and the basis for screening or including the individual pathways. Additional pathways were identified that should be included in the Fire PRA model. However, these pathways appear to not be included under gate 1(2)-CONTISOLATION." The licensee states that this is only a documentation upgrade and that there is no impact on PRA quantification.*

According to Table 4-2.5-3(b) in ASME/ANS RA-Sa-2009, PRM-B15, "all the SRs under HLR-LE-A, HLR-LE-B, HLR-LE-C, and HLR-LE-D in Part 2 are to be addressed in the context of fire scenarios including effects on system operability/functionality, operator actions, accident progression, and possible containment failures accounting for fire damage to equipment and associated cabling."

F&O No. 15-15 indicates that the issue requires more than a documentation only upgrade. Correct the model by including these missing pathways under gate "1(2)-CONTISOLATION" and update the quantification results, or justify why these pathways have no significant impact on the quantification results used in this application.

EGC Response to RAI-17c:

PRM-B15 was resolved by incorporating logic for containment isolation valves into the LERF logic for the Fire PRA. The resolution to this F&O was incorporated in the model used for quantification of results for the initial LAR submittal. A review of the remaining documentation-only F&O dispositions was performed and it was confirmed there were no dispositions associated with changes which could impact the results documented in this LAR. Table 4.4 of Attachment 2 was revised to reflect this resolution.

REFERENCES

1. Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request for a One-Time Extension of the Essential Service Water (SX) Train Completion Time to Support 2A SX Pump Repair," dated September 30, 2016 (ML16274A474)

2. Email from J. Wiebe (NRC) to J. Bauer (Exelon Generation Company, LLC), "Initial RAIs Related to the Braidwood Unit 2 SX Pump allowed Outage Time Amendment Request," dated October 20, 2016

3. Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information License Amendment Request for a One-Time Extension of the Essential Service Water (SX) Train Completion Time to Support 2A SX Pump Repair," dated October 26, 2016

4. Email from J. Wiebe (NRC) to J. Bauer (Exelon Generation Company, LLC), "Supplemental Initial RAIs Related to the Braidwood Unit 2 SX Pump allowed Outage Time Amendment Request," dated October 22, 2016

ATTACHMENT 2

BW-LAR-008, "Risk Assessment Input for the Braidwood One-Time Technical Specification Change for the Essential Service Water Pump 2A Completion Time from 72 to 200 Hours," Revision 1

Braidwood

PRA APPLICATION NOTEBOOK

BW-LAR-008

Risk Assessment Input for the Braidwood One-Time Technical Specification Change for the Essential Service Water Pump 2A Completion Time from 72 Hours to 200 Hours

REVISION 1

RM DOCUMENTATION NO: BW-LAR-008	REV: 1	PAGE NO. 2

STATION: Braidwood

UNIT(S) AFFECTED: Unit 2

TITLE: Risk Assessment Input for the Braidwood One-Time Technical Specification Change for the Essential Service Water Pump 2A Completion Time from 72 Hours to 200 Hours

SUMMARY: This assessment is performed in support of the License Amendment Request (LAR) submittal for a one-time change to extend the Completion Time (CT) for the Unit 2 Essential Service Water A pump from 72 hours to 200 hours in order to allow for repairs of the pump train. See the compensatory measures in Section 5.4.1 that support this risk analysis and constitute good risk management actions for this 2A SX outage.

~~The risk assessment is performed in accordance with ER-AA-600-1046, Rev. 6, Risk Metrics –~~ NOED and LAR [Ref. 19].

[ ] Review required after periodic Update

[X] Internal RM Documentation **[ ] External RM Documentation**

Electronic Calculation Data Files: BW-LAR-008-Attachments.zip/8.149MB/9-26-16/11:02AM

Method of Review: [X] Detailed [ ] Alternate[ ] Review of External Document

This RM documentation supersedes: N/A in its entirety.

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TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to assess the acceptability, from a risk perspective, of a change to extend the Braidwood Station completion time (CT) for the 2A Essential Service Water (SX) Pump Train in Tech Spec (TS) 3.7.8 from 72 hours to 200 hours in order to allow for replacement of the pump. These proposed changes are requested to be effective only during a one-time outage in the Fall of 2016.

The analysis follows the guidance provided in Regulatory Guide 1.200 Revision 2 [Ref. 1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

1.2 BACKGROUND

1.2.1 Technical Specification Changes

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it . . .

> . . . expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements.
3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
4. The Commission's safety goals and subsidiary numerical objectives are to be used with consideration of uncertainties in making regulatory judgments...

The movement of the NRC to more risk-informed regulation has led to the NRC identifying Regulatory Guides and associated processes by which licensees can submit changes to the plant design basis including Technical Specifications. Regulatory Guides 1.174 [Ref. 2] and 1.177 [Ref. 3] both provide processes to incorporate PRA input for decision makers regarding a Technical Specification modification.

1.3 REGULATORY GUIDES

Three Regulatory Guides provide primary inputs to the evaluation of a Technical Specification change. Their relevance is discussed in this section.

1.3.1 Regulatory Guide 1.200, Revision 2

Regulatory Guide 1.200, Revision 2 [Ref. 1] describes an 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 decision-making for light-water reactors. This guidance is

intended to be consistent with the NRC's PRA Policy Statement and more detailed guidance in Regulatory Guide 1.174.

It is noted that RG 1.200, Revision 2 endorses Addendum A of the ASME/ANS PRA Standard [Ref. 5] as clarified in Appendix A of RG 1.200, Revision 2.

1.3.2 Regulatory Guide 1.174, Revision 2

Regulatory Guide 1.174 [Ref. 2] specifies an approach and acceptance guidelines for use of PRA in risk informed activities. RG 1.174 outlines PRA related acceptance guidelines for use of PRA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for the evaluation of permanent TS changes. The guidelines given in RG 1.174 for determining what constitutes an acceptable permanent change specify that the ΔCDF and the ΔLERF associated with the change should be less than specified values, which are dependent on the baseline CDF and LERF, respectively.

RG 1.174 also specifies guidelines for consideration of external events. External events can be evaluated in either a qualitative or quantitative manner.

Since this LAR is for a one-time TS change, the ΔCDF and the ΔLERF of RG 1.1.74 do not specifically apply.

1.3.3 Regulatory Guide 1.177 Revision 1

Regulatory Guide 1.177 [Ref. 3] specifies an approach and acceptance guidelines for the evaluation of plant licensing basis changes. RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change as identified below:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in core damage frequency (CDF) and incremental conditional core damage probability

(ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of large early release frequency (LERF) and incremental conditional large early release probability (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable <u>permanent</u> TS change is that the ICCDP and the ICLERP associated with the change should be less than 1E-06 and 1E-07, respectively. RG 1.177 also addresses risk metric requirements for one-time TS changes, as outlined in Section 1.3.4 of this risk assessment.

- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.
- Tier 3 provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the CT. Compared with Tier 2, Tier 3 provides additional coverage based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 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.

This risk analysis supports the Tier 1 element of RG 1.177, specifically the comparison of the results with the acceptance guidelines for ICCDP and ICLERP associated with

changing a Technical Specification Completion Time. Other portions of the LAR submittal will address Tier 2 and Tier 3 elements.

1.3.4 Acceptance Guidelines

Risk significance in an LAR is determined by comparison of changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) and values of Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) produced by a permanent change to either the plant design basis or Technical Specifications to the guidelines given in Regulatory Guide 1.174 and Regulatory Guide 1.177. Reg. Guide 1.174 specifies the acceptable changes in CDF and LERF for permanent changes. Reg. Guide 1.177 specifies the acceptable ICCDP and ICLERP for permanent changes, usually associated with changing CT.

Also Reg. Guide 1.177 directly addresses the risk metric requirements for one-time TS changes, as reproduced below:

> *"For one-time only changes to TS CTs, the frequency of entry into the CT may be known, and the configuration of the plant SSCs may be established. Further, there is no permanent change to the plant CDF or LERF, and hence the risk guidelines of Regulatory Guide 1.174 cannot be applied directly. The following TS acceptance guidelines specific to one-time only CT changes are provided 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):*
>   - *ICCDP of less than $1.0x10^{-6}$ and an ICLERP of less than $1.0x10^{-7}$,or*
>   - *ICCDP of less than $1.0x10^{-5}$ and an ICLERP of less than $1.0x10^{-6}$ 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)."*

Based on the available quantitative guidelines for other risk-informed applications, it is judged that the quantitative criteria shown in Table 1-1 represent a reasonable set of acceptance guidelines. For the purposes of this evaluation, these guidelines demonstrate that the risk impacts are acceptably low. This combined with effective compensatory measures to maintain lower risk will ensure that the TS change meets the intent of small risk increases consistent with the Commission's Safety Goal Policy Statement.

Table 1-1
PROPOSED RISK ACCEPTANCE GUIDELINES

RISK ACCEPTANCE GUIDELINE	BASIS
ICCDP < 1E-6, or ICCDP < 1E-5 with effective compensatory measures implemented to reduce the sources of increased risk	ICCDP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.
ICLERP < 1E-7, or ICLERP < 1E-6 with effective compensatory measures implemented to reduce the sources of increased risk	ICLERP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.

1.4 SCOPE

This section addresses the requirements of RG 1.200, Revision 2 Section 3.1 which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors are as follows:

- Full Power Internal Events (FPIE) – The Braidwood PRA model used for this analysis includes a full range of internal initiating events (including internal flooding) for at-power configurations. The Essential Service Water system (SX) is credited in the PRA for support of cooling for Safety-Related cooling loads including the emergency diesel generators (DG), component cooling water (CC), auxiliary feedwater (AF), and various room coolers. The FPIE model is further discussed in Section 1.5.
- Low Power Operation - The FPIE assessment is judged to adequately capture risk contributors associated with low power plant operations. The FPIE analysis assumes that the plant is at full power at the time of any internal events transient, manual shutdown, or accident initiating event. This analytic approach results in conservative accident progression timings and systemic success criteria compared to what may otherwise be applicable to an initiator occurring at low power. As such, low power risk impacts are not discussed further in this risk assessment.
- Shutdown / Refueling – Braidwood does not have a shutdown PRA model, but instead relies upon deterministic methodology to assess defense-in-depth of key safety functions. The intent is for the unit to remain at-power for the duration of the extended CT.
- Internal Fires –Braidwood currently has an interim, peer-reviewed fire PRA model. The Braidwood working Fire PRA [Ref. 10] is used to provide both quantitative and qualitative insights to the analysis of the 2A SX CT extension (refer to Section 3.3.2).
- Seismic - Braidwood does not currently maintain a Seismic PRA. A qualitative assessment is performed in this analysis (refer to Section 3.3.3).
- High Winds – Braidwood does not have a high winds or tornado PRA. A qualitative assessment is performed in this analysis (refer to Section 3.3.4).

- External Flood – Braidwood does not have an external flood PRA. A qualitative assessment is performed in this analysis (refer to Section 3.3.5).
- Other External Events - Other external event risks were assessed in the Braidwood IPEEE study [Ref. 13] and found to be insignificant risk contributors. These conclusions are revisited for this 2A SX CT extension assessment (refer to Section 3.3.6).

1.5 BRAIDWOOD PRA MODELS

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 2 [Ref. 1] which directs the licensee to identify the portions of the PRA used in the analysis.

The PRA analysis uses the BB011b4 full power internal events (FPIE) Level 1 Core Damage Frequency (CDF) model and the associated Level 2 Large Early Release Frequency (LERF) model to calculate the risk metrics [Ref. 7]. The PRA analysis also uses the fire model BB011b-FL-B [Ref. 10] to calculate the risk metrics for full power internal fires to develop quantitative and qualitative risk insights. Section 3.2 details the internal events analysis using the FPIE PRA, and Section 3.3 details the fire risk assessment.

2.0 ANALYSIS ROADMAP AND REPORT ORGANIZATION

The analysis and documentation utilizes the guidance provided in RG 1.200, Revision 2. The guidance in RG 1.200, Revision 2 indicates that the following steps should be followed to perform this study:

1. Per Section 3. of RG 1.200, include the following information regarding the PRA to support the application
    a. Describe the SSCs, operator actions, and operational characteristics affected by the application and how these are implemented in the PRA model.
    b. Provide a definition of the acceptance guidelines used for the application.

2. Per Section 3.1 of RG 1.200, identify the scope of risk contributors addressed by the PRA model
    a. If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.

3. Per Section 3.2 of RG 1.200, identify the parts of the PRA used to support the application
    a. Identify the logic model elements onto which the relevant SSCs, operator actions, and operational characteristics are mapped to the PRA model.
    b. Identify the relevant accident sequences that are impacted by the changes identified in the first group.

4. Per Section 3.3 and 4.2 of RG 1.200, demonstrate the Technical Adequacy of the PRA

a. Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.

b. Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.

c. Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.

d. Identify key assumptions and approximations relevant to the results used in the decision-making process.

5. Per Section 4.2 of RG 1.200, summarize the risk assessment methodology used to assess the risk of the application

    a. Include how the PRA model was modified to appropriately model the risk impact of the change request.

Table 2-1 summarizes the RG 1.200 identified actions and the corresponding location of that analysis or information in this report.

Table 2-1
RG 1.200 ANALYSIS ACTIONS ROADMAP

RG 1.200 Actions	Report Section
1a. Describe the SSCs, operator actions, and operational characteristics affected by the application and how these are implemented in the PRA model.	Section 1.5 and Section 3.1.1
1b. Provide a definition of the acceptance guidelines used for the application.	Section 1.3.4
2. Identify the scope of risk contributors addressed by the PRA model.	Section 1.4
2a. If not full scope (i.e., internal and external events), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.	Section 3.3
3. Identify the parts of the PRA used to support the application	Section 1.5 and Section 3
3a. Identify logic model elements that are mapped to the PRA model	Section 3.1 and Section 3.2
3b. Identify the accident sequences impacted by those changes.	Section 3
4. Demonstrate the Technical Adequacy of the PRA.	Section 4
4a. Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.	Section 4.6.1, Table 4-1
4b. Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the RG. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.	Section 4.6.2, Table 4-2
4c. Document PRA peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.	Section 4.6.3, Table 4-3
4d. Identify key assumptions and approximations relevant to the results used in the decision-making process.	Section 3.1 and Section 3.5
5. Summarize the risk assessment methodology used to assess the risk of the application. Include how the PRA model was modified to appropriately model the risk impact of the change request.	Section 1.5 and Section 3

3.0 RISK ANALYSIS

This section evaluates the plant-specific risk associated with the proposed TS change, based on the risk metrics of CDF, ICCDP, LERF, and ICLERP.

3.1 ASSESSMENT OVERVIEW AND ASSUMPTIONS

3.1.1 Overview

This analysis is performed for unavailability of the 2A SX pump. The PRA analysis involves identifying the system and components or maintenance activities modeled in the PRA which are most appropriate for use in representing the extended CT configurations and comparing the results to the baseline. The base risk metrics for the FPIE PRA and the FPRA are established in Table 3.1-1.

Table 3.1-1

BRAIDWOOD FPIE PRA CDF AND LERF BASE RISK METRICS

Risk	**BB011b4- Unit 1 (/yr)**	**BB011b4- Unit 2 (/yr)**
FPIE CDF	1.96E-5	1.94E-5
FPIE LERF	9.60E-7	9.52E-7
Risk Metric	**BB011b4- Unit 1 (/yr)**	**BB011b4- Unit 2 (/yr)**
Fire CDF	4.25E-5	5.32E-5
Fire LERF	7.73E-6	5.96E-6

The general configuration for the extended CT is Braidwood at-power on both units with the 2A SX pump train out of service. The planned maintenance is expected to focus on repair of the rotating element/impeller of the pump with a contingency to replace the entire pump within the requested extended CT. The pump maintenance will be done in a workweek where the pump maintenance will be the focus of the week and there will not be significant concurrent maintenance work. The opposite division train (2B) and the Unit 1 SX trains (1A and 1B) will be protected. Additionally, all station emergency

diesel generators will also be protected. A complete list of protected equipment and other compensatory measures is discussed in Section 5.4.1.

Initially, the PRA model was quantified using the base "average test and maintenance" PRA model with the 2A SX out for maintenance. The average test and maintenance model represents baseline assumed maintenance frequencies for all components with the exception of Technical Specification violations that are normally excluded in the disallowed maintenance logic in the base PRA model. After analyzing the risk results, other maintenance terms became candidates for restricting elective maintenance to help reduce the overall risk associated with the extended CT. Restricted maintenance is discussed further in Section 3.1.2. This configuration is represented in the PRA as shown below in Table 3.1-2.

Table 3.1-2
2A SX PUMP EXTENDED CT CONFIGURATION REPRESENTATION

BASIC EVENT	DESCRIPTION	VALUE
2SX01PA-----PMMM	SX PUMP 2A UNAVAILABLE DUE TO MAINTENANCE	TRUE
FLAG-SX-PUMP-2B	SX PUMP 2B IS IN STANDBY (FLAG)	FALSE(1)
FLAG-SX-PUMP-2A	SX PUMP 2A IS IN STANDBY (FLAG)	TRUE(1)
2SX01PB-----PMMM	SX PUMP 2B UNAVAILABLE DUE TO MAINTENANCE	FALSE(2)
1SX01PA-----PMMM	SX PUMP 1A UNAVAILABLE DUE TO MAINTENANCE	FALSE(2)
1SX01PB-----PMMM	SX PUMP 1B UNAVAILABLE DUE TO MAINTENANCE	FALSE(2)
2DG2A-------DGMM	DIESEL GENERATOR 2A UNAVAILABLE DUE TO MAINTENANCE AT POWER	FALSE(3)
2DG2B-------DGMM	DIESEL GENERATOR 2B UNAVAILABLE DUE TO MAINTENANCE AT POWER	FALSE(3)
1DG1A-------DGMM	DIESEL GENERATOR 1A UNAVAILABLE DUE TO MAINTENANCE AT POWER	FALSE(3)
1DG1B-------DGMM	DIESEL GENERATOR 1B UNAVAILABLE DUE TO MAINTENANCE AT POWER	FALSE(3)
2AF01PA-----PMMM	AF MOTOR-DRIVEN PUMP 2AF01PA UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2AF01PB-----PDMM	AF DIESEL-DRIVEN PUMP 2AF01PB UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
1SX016A027A-MVMM	SX PUMP A MIN FLOW PATH VIA SX 16A\ 27A (RCFC 1A\1C) IS ISOLATED	FALSE(3)
1SX016B027B-MVMM	SX PUMP B MIN FLOW PATH VIA SX 16B\ 27B (RCFC 1B\1D) IS ISOLATED	FALSE(3)

BASIC EVENT	DESCRIPTION	VALUE
2SX016A027A-MVMM	SX PUMP A MIN FLOW PATH VIA SX 16A\ 27A (RCFC 2A\2C) IS ISOLATED	FALSE(3)
2SX016B027B-MVMM	SX PUMP 2B MIN FLOW PATH VIA SX 16B\ 27B (RCFC 2B\2D) IS ISOLATED	FALSE(3)
2IP211------IXMM	INVERTER 211 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP212------IXMM	INVERTER 212 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP213------IXMM	INVERTER 213 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP214------IXMM	INVERTER 214 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP211------TRMM	480-120V TRANSFORMER 211 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP212------TRMM	480-120V TRANSFORMER 212 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP213------TRMM	480-120V TRANSFORMER 213 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)
2IP214------TRMM	480-120V TRANSFORMER 214 UNAVAILABLE DUE TO MAINTENANCE	FALSE(3)

Notes to Table 3.1-2:

(1) The Braidwood PRA model BB011b4 uses model flags to set the default alignment of the SX pumps. Setting the flag to TRUE places the pump in standby, while setting the flag to FALSE indicates to the model that the pump should be running.

(2) During the proposed maintenance of the 2A SX, there is to be no elective maintenance of the 1A, 1B, or 2B SX pumps.

(3) These terms represent additional elective maintenance that will be restricted during the execution of the 2A SX extended CT.

3.1.2 Assumptions

The following assumptions are used in quantifying the plant risk due to the one-time 2A SX Pump CT.

- The 2A SX 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 FPIE PRA analysis assumes unavailability of the 2A SX pump via its corresponding maintenance basic event.
- No elective maintenance will be performed on the SX 1A, 1B, 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 auxiliary feed (AF) pumps. These maintenance terms are set to FALSE for the quantification.
- There will be no elective maintenance on the SX 16A/B or 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.

3.2 INTERNAL EVENTS

3.2.1 FPIE PRA Evaluation and Results

The proposed technical specification change involves unavailability of the 2A SX pump. The revised CDF and LERF values for the CT configurations are obtained by re-quantifying the base PRA model with all of the identified events set as shown in Table 3.1-2. The 2A SX maintenance term was set to TRUE using a flag file, while disallowed maintenance terms were set to FALSE.

The evaluation of ICCDP and ICLERP for the 2A SX CT change is determined as shown below:

The ICCDP associated with 2A SX pump being OOS using the new CT is given by

$$ICCDP_{2A\ SX} = (CDF_{2A\ SX} - CDF_{BASE}) \times CT_{NEW} \qquad [\textit{Eq. 3-1}]$$

where

$CDF_{2A\ SX}$ = the annual average CDF calculated with the 2A SX equipment OOS

CDF_{BASE} = baseline annual average CDF with average unavailability for all equipment. This is the CDF result of the baseline PRA.

CT_{NEW} = the new extended CT (in units of hours, e.g. 200 hours * 1 year / 365 days / 24 hours = 2.28E-2 years)

Note: ICCDP is a dimensionless probability.

Risk significance relative ICLERP is determined using equations of the same form as noted above for ICCDP.

Since this evaluation is for a one-time Tech Spec CT allowance, the ICCDP and ICLERP are the only meaningful metrics as there is no permanent change in plant risk after this one-time CT extension.

The relevant inputs to Equation 3-1 (and the equivalent for LERF) are shown in Table 3.2-1 below. The corresponding output parameters from the equations above are then provided in Table 3.2-2. The analysis is performed for CDF and LERF from the internal events and internal floods Unit 2 PRA model.

Table 3.2-1
FPIE Risk Assessment Input Parameters and Results for Unit 2

Input Parameter	Value
CDF_{BASE}	1.94E-05/yr(1)
$CDF_{2A\ SX}$	2.25E-05/yr(1)
$LERF_{BASE}$	9.52E-07/yr(2)
$LERF_{2A\ SX}$	1.00E-06/yr(2)
CT_{NEW}	2.28E-02 yrs

(1) Based on a truncation of 1E-10

(2) Based on a truncation of 1E-11

Table 3.2-2
FPIE PRA Risk Assessment Base Output Results for Unit 2

Risk Metric	Value
$ICCDP_{2A\ SX}$	7.2E-08
$ICLERP_{2A\ SX}$	1.1E-09

In addition to the CDF/LERF calculations, a sequence review is performed as directed by ER-AA-600-1046 [Ref. 19]. This analysis consists of determining if significant changes to accident sequences exist due to the extended CT configuration.

As shown in Table 3.2-3, for 2A SX OOS, general transient sequences contribute 41.5% of the risk associated with the 2A SX Pump OOS sequence quantification. Sequences involving small loss of coolant accidents (SLOCA) contribute 30.1% of the risk associated with the 2A SX Pump OOS sequence quantification, followed by loss of offsite power (LOOP) sequences at 11.1%. These results indicate that the Transient (reactor trip), SLOCA and LOOP sequences dominate the source of station risk for the configuration with the 2A SX pump being out-of-service. In comparison to the base case results, the majority of the sequence results (with the exception of LOOP sequences) were similar to the base case results.

Table 3.2-3: Comparison of Sequence Contributions for the 2A SX Pump OOS Case

Sequence Group	2A SX OOS CDF	% Contribution	Base Case Contribution
2TRAN	1.02E-05	41.5%	42.5%
2SLOC	7.41E-06	30.1%	30.5%
2LOOP	2.74E-06	11.1%	6.1%
2SGTR	1.68E-06	6.8%	7.9%
2MLOC	1.48E-06	6.0%	7.0%
2ILOC	3.40E-07	1.4%	1.6%
2SLBI	2.21E-07	0.9%	1.0%
2LODC	2.06E-07	0.8%	1.8%
2ATWS	1.60E-07	0.6%	0.8%
2SLBO	1.53E-07	0.6%	0.7%
2LLOC	1.20E-08	0.0%	0.1%
2XLOC	2.32E-09	0.0%	0.0%

The second aspect of the 2A SX risk characterization that can be taken from the internal events model is the type of initiating event(s) that contribute to the CDF associated with 2A SX being out of service. As shown in Table 3.2-4, the largest contribution comes from a loss of SX, followed by SLOCA and loss of offsite power events. Note that the Loss of SX, Loss of Component Cooling Water, and Loss of AC Power (AP) (non-LOOP) events are evaluated through the Transient event trees for sequence quantification. These results are consistent with the results of the sequence analysis. These insights indicate that transient/reactor trip, SLOCA, LOOP, and other initiators all have the potential to create a demand for SX.

Table 3.2-4: CDF Contribution by Initiating Event Group

Initiating Event Group	% Contribution	Base Case Contribution
Loss of SX	34.3%	37.5%
Small LOCA	13.4%	16.1%
LOOP & DLOOP	11.4%	2.0%
SGTR	7.1%	8.6%
Internal Flooding	6.8%	7.7%
Loss of AP	6.7%	1.4%
Loss of CCW	6.6%	8.0%
Medium LOCA	6.4%	7.6%
Other	5.0%	7.0%
General Transient & LMFW	2.3%	3.9%

In addition, the dominant cutsets were reviewed. The top 20 cutsets (representing ~50% of CDF contribution) for the 2A SX unavailable configuration are shown in Table 3.2-5

Table 3.2-5: Top 20 CDF Cutsets for the 2A SX Pump OOS Configuration

Cutset #	Cutset Prob.	Event Prob	Event	Event Description
1	2.90E-06	9.60E-01	%SXIE	INDICATOR FOR SX INITIATING EVENT
		2.16E-04	0SX01AB2AB-CPMFRIE	FAILURE OF ALL SX PUMPS (1A/1B/2A/2B) TO RUN DUE TO CCF (4/4)
		1.40E-02	2FW-FWR---EHSYOA	OPERATORS FAIL TO EXECUTE FW RESTORATION
2	1.29E-06	9.60E-01	%SXIE	INDICATOR FOR SX INITIATING EVENT
		2.16E-04	0SX01AB2AB-CPMFRIE	FAILURE OF ALL SX PUMPS (1A/1B/2A/2B) TO RUN DUE TO CCF (4/4)
		6.25E-03	2AP-BOTHSAT-TRMM	BOTH U2 SAT OOS FOR TM - 241 PWR VIA 141; 242 PWR VIA 142; 256 - 259 ON UAT
3	1.19E-06	6.98E-04	%RC-SLOC2-N-PSIE	SMALL LOCA INITIATING EVENT (NON-ISOLABLE)
		3.40E-03	0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS
		5.00E-01	FLAG-CCHTX0-U1	CCW HTX 0 ALIGNED TO UNIT 1
4	1.19E-06	6.98E-04	%RC-SLOC2-N-PSIE	SMALL LOCA INITIATING EVENT (NON-ISOLABLE)
		3.40E-03	0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS
		5.00E-01	FLAG-CCHTX0-U2	CCW HTX 0 ALIGNED TO UNIT 2
5	6.78E-07	3.99E-04	%RC-MLOC2---PMIE	MEDIUM LOCA INITIATING EVENT
		3.40E-03	0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS
		5.00E-01	FLAG-CCHTX0-U1	CCW HTX 0 ALIGNED TO UNIT 1
6	6.78E-07	3.99E-04	%RC-MLOC2---PMIE	MEDIUM LOCA INITIATING EVENT

Table 3.2-5: Top 20 CDF Cutsets for the 2A SX Pump OOS Configuration

Cutset #	Cutset Prob.	Event Prob	Event	Event Description
		3.40E-03	0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS
		5.00E-01	FLAG-CCHTX0-U2	CCW HTX 0 ALIGNED TO UNIT 2
7	5.95E-07	4.23E-04	%FL2WSM3A0----T1	UNIT 2 MAJOR FLOOD (>3,700GPM) FROM NORMAL SERVICE WATER INTO AUX BLDG - COMMON
		3.90E-03	FLMITIG-M3-T1-WS	
		3.60E-01	2FP-PRI-7D-HMVRA	RECOV OF LOSS OF SX SEAL LOCA (COND PROB OF 2FP-PRI-7D-HMVRA + 0.21 SEAL FAIL)
8	5.10E-07	6.98E-04	%RC-SLOC2-N-PSIE	SMALL LOCA INITIATING EVENT (NON-ISOLABLE)
		7.30E-04	2RH-SP-X---HPMOA	OPERATORS FAIL TO STOP RH PUMPS
9	4.24E-07	9.60E-01	%APIE	INDICATOR FOR AP INITIATING EVENT
		2.21E-03	2AP242------BSLPIE	BUS 242 FAILS
		2.00E-04	2RX-JHEP26-HOADA	JOINT HEP FOR 0SX-XTIE---HMVOA AND 2AF-AF005--HAVOA
10	3.33E-07	9.60E-01	%SXIE	INDICATOR FOR SX INITIATING EVENT
		2.16E-04	0SX01AB2AB-CPMFRIE	FAILURE OF ALL SX PUMPS (1A/1B/2A/2B) TO RUN DUE TO CCF (4/4)
		2.40E-03	0AP-DLOOP-GT	CONDITIONAL PROBABILITY OF DLOOP GIVEN GENERAL TRANSIENT
		6.70E-01	0AP-DLOOP-SC	FRACTION OF CONDITIONAL LOOPS THAT ARE SWITCHYARD-CENTERED
11	2.78E-07	8.41E-04	%RC-SGTR2-A-HXIE	STEAM GENERATOR TUBE RUPTURE IN S/G 2A
		3.30E-04	2RX-JHEP28-HOADA	JOINT HEP FOR 2RC-DS-SGTRHDVOA AND 2RC-LCD----HSYOA

Table 3.2-5: Top 20 CDF Cutsets for the 2A SX Pump OOS Configuration

Cutset #	Cutset Prob.	Event Prob	Event	Event Description
12	2.78E-07	8.41E-04	%RC-SGTR2-B-HXIE	STEAM GENERATOR TUBE RUPTURE IN S/G 2B
		3.30E-04	2RX-JHEP28-HOADA	JOINT HEP FOR 2RC-DS-SGTRHDVOA AND 2RC-LCD----HSYOA
13	2.78E-07	8.41E-04	%RC-SGTR2-C-HXIE	STEAM GENERATOR TUBE RUPTURE IN S/G 2C
		3.30E-04	2RX-JHEP28-HOADA	JOINT HEP FOR 2RC-DS-SGTRHDVOA AND 2RC-LCD----HSYOA
14	2.78E-07	8.41E-04	%RC-SGTR2-D-HXIE	STEAM GENERATOR TUBE RUPTURE IN S/G 2D
		3.30E-04	2RX-JHEP28-HOADA	JOINT HEP FOR 2RC-DS-SGTRHDVOA AND 2RC-LCD----HSYOA
15	2.57E-07	9.16E-07	%RCS-RHR-DISCHIE	FREQ OF EXPOSING RHR PUMP DISCHARGE HEADERS TO RCS PRESSURE
		2.80E-01	LEAK-800-150	CONDITIONAL PROB OF LEAK 800 GPM GIVEN LEAK IS AT LEAST 150 GPM
16	2.55E-07	9.60E-01	%CCIE	INDICATOR FOR CC INITIATING EVENT
		4.96E-04	2CC01PA-B--CPMFRIE	CCW PUMPS 2CC01PA & 2CC01PB FAIL TO RUN DUE TO CCF (2/4)
		5.10E-03	2RX-JHEP87-HOADA	JOINT HEP FOR 2CV-ALL----HPMOA AND 2RC-PMTRIPAHSYOA
		5.00E-01	FLAG-CCHTX0-U1	CCW HTX 0 ALIGNED TO UNIT 1
		2.10E-01	SEAL-U2-TRANS	UNIT 2 SEAL LOCA >21GPM RANDOMLY OCCURS - NON-LOOP SEQUENCES
17	2.55E-07	9.60E-01	%CCIE	INDICATOR FOR CC INITIATING EVENT
		4.96E-04	2CC01PA-B--CPMFRIE	CCW PUMPS 2CC01PA & 2CC01PB FAIL TO RUN DUE TO CCF (2/4)
		5.10E-03	2RX-JHEP87-HOADA	JOINT HEP FOR 2CV-ALL----HPMOA AND 2RC-PMTRIPAHSYOA

Table 3.2-5: Top 20 CDF Cutsets for the 2A SX Pump OOS Configuration

Cutset #	Cutset Prob.	Event Prob	Event	Event Description
		5.00E-01	FLAG-CCHTX0-U2	CCW HTX 0 ALIGNED TO UNIT 2
		2.10E-01	SEAL-U2-TRANS	UNIT 2 SEAL LOCA >21GPM RANDOMLY OCCURS - NON-LOOP SEQUENCES
18	2.28E-07	9.60E-01	%SXIE	INDICATOR FOR SX INITIATING EVENT
		2.16E-04	0SX01AB2AB-CPMFRIE	FAILURE OF ALL SX PUMPS (1A/1B/2A/2B) TO RUN DUE TO CCF (4/4)
		1.10E-03	2FW-FRH1---HSGOA	OPERATORS FAIL RECOGNIZE THE CUE TO SECONDARY COOLING
19	2.12E-07	9.60E-01	%APIE	INDICATOR FOR AP INITIATING EVENT
		2.21E-03	2AP242------BSLPIE	BUS 242 FAILS
		2.00E-04	2RX-JHEP76-HOADA	JOINT HEP FOR 0SX005-----HMVOA AND 2AF-AF005--HAVOA
		5.00E-01	FLAG-CCHTX0-U1	CCW HTX 0 ALIGNED TO UNIT 1
20	2.11E-07	9.60E-01	%SXIE	INDICATOR FOR SX INITIATING EVENT
		1.57E-05	0SX-ALL----CSRPGIE	SX STRAINERS - PLUGGED DUE TO CCF (4/4)
		1.40E-02	2FW-FWR---EHSYOA	OPERATORS FAIL TO EXECUTE FW RESTORATION

Consistent with the contribution identified in Table 3.2-1 by Sequence and Table 3.2-2 by Initiating Event, the top cutsets involve loss of SX, Loss of AP and Small LOCA events. A further review of the cutsets from Table 3.2-5 identifies the following operator actions as being important to the assessment, shown in Table 3.2-6.

Table 3.2-6: Significant Operator Actions From Cutset Reviews

Basic Event	Description
2FW-FWR---EHSYOA	OPERATORS FAIL TO EXECUTE FW RESTORATION
0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS

Operating Crew briefings to identify and review these actions for the duration of the extended CT would be prudent. In addition to the operator actions, one maintenance unavailability event associated with the U2 SATs was also identified. A recommendation to the station to restrict any elective maintenance on the Unit 2 SATs as a compensatory action has been made though this action has not been incorporated into the PRA results.

Table 3.2-7 provides a review of basic event importance for the 2A SX pump unavailability case. This table shows basic events with more than 1% contribution to CDF. For this table, FLAG events, alignment events, initiating events and so forth were excluded from this list.

Table 3.2-7: Basic Events with Greater than 1% CDF Contribution

Event	Description	FV - CDF
0SX007-ES13HMVOA	OPERATORS FAIL TO LOCALLY THROTTLE SX007 TO CC HXS	1.83E-01
2FW-FWR---EHSYOA	OPERATORS FAIL TO EXECUTE FW RESTORATION	1.45E-01
2RX-JHEP87-HOADA	JOINT HEP FOR 2CV-ALL----HPMOA AND 2RC-PMTRIPAHSYOA	1.01E-01
2AP-BOTHSAT-TRMM	BOTH U2 SAT OOS FOR TM - 241 PWR VIA 141; 242 PWR VIA 142; 256 - 259 ON UAT	6.43E-02
2RX-JHEP28-HOADA	JOINT HEP FOR 2RC-DS-SGTRHDVOA AND 2RC-LCD----HSYOA	4.72E-02

Table 3.2-7: Basic Events with Greater than 1% CDF Contribution

Event	Description	FV - CDF
0SX-XTIE-D-HMVRA	RECOV OF LOSS OF SX SEAL LOCA (COND PROB OF 0SX-XTIE-D-HMVRA + 0.21 SEAL FAIL)	3.36E-02
2RC-PMTRIPAHSYOA	OPS FAIL TO TRIP RCP TO PROTECT SDS	3.29E-02
2RX-JHEP22-HOADA	JOINT HEP FOR 0SX-XTIE---HMVOA AND (2FP-PRI-7X-HMVOA OR 2CV-ALL----HPMOA)	3.12E-02
2DG2B-------DGMM	DIESEL GENERATOR 2B UNAVAILABLE DUE TO MAINTENANCE AT POWER	2.93E-02
2FP-PRI-7D-HMVRA	RECOV OF LOSS OF SX SEAL LOCA (COND PROB OF 2FP-PRI-7D-HMVRA + 0.21 SEAL FAIL)	2.90E-02
FLMITIG-M3-T1-WS	FAILURE TO MITIGATE >3700 WS FLOOD FOR T1 SCENARIO	2.79E-02
2CV-ALL-D--HPMRA	RECOV OF LOSS OF SX SEAL LOCA (COND PROB OF 2CV-ALL-D-HPMRA + 0.21 SEAL FAIL)	2.55E-02
2RH-SP-X---HPMOA	OPERATORS FAIL TO STOP RH PUMPS	2.54E-02
2DG2B-------DGFS	DG 2B FAILS TO START RANDOMLY	2.41E-02
2RX-JHEP26-HOADA	JOINT HEP FOR 0SX-XTIE---HMVOA AND 2AF-AF005--HAVOA	1.95E-02
2FW-FRH1---HSGOA	OPERATORS FAIL RECOGNIZE THE CUE TO SECONDARY COOLING	1.74E-02
2AP-XTE-SBOHHBOA	POWER XTIE FAILS EARLY (AF FAILED TIME TO XTIE <30 MIN) SBO XTIE	1.58E-02
0SX-XTIE---HMVOA	OPERATORS FAIL TO OPEN SX UNIT XTIE VALVES	1.46E-02
0RC-SDSFAIL-SLOO	FAILURE OF THE SDS TO ACTUATE AND INITIALLY SEAL	1.44E-02
2AF01PA-B--CPMFR	AF PUMPS FAIL TO RUN DUE TO CCF (2/2)	1.40E-02
2FW02P------PMMM	MFW MD START UP PUMP FW02P UNAVAILABLE DUE TO MAINTENANCE	1.38E-02
2AP-XTL-SBOHHBOA	XTIE FAILS FOR SEQUENCES WITH AF AVAIL FOR FIRST FOUR HOURS (SBO XTIE)	1.37E-02
2DG2B-------DGFR	DG 2B FAILS TO RUN	1.21E-02
2AP242SQCMB-SQMF	SEQUENCER FAILS IN A MANNER THAT FAILS THE DIESEL	1.04E-02

The basic events with the largest contribution (>10% each) are associated with operator actions. The top two basic events have been identified through the cutset review performed above. The third action is a combination HEP event that consists primarily of the action to trip the RCPs to prevent damage to the shutdown seals. The basic events associated with the capability of the 2B DG to start and run comprise ~5% contribution to CDF. From this review, the following items are identified:

- Operating Crew briefings to review the action to trip the RCPs to preclude damage to the Shutdown Seals for the duration of the extended CT would be prudent.
- Prohibit planned maintenance activities on the 2B DG for the duration of the extended CT condition. Note that this action is credited in the 2A SX pump unavailable quantification.
- Prohibit planned maintenance activities on the motor-driven startup Feedwater pump, 2FW02P, for the duration of the extended CT. Note that this action is not credited in the 2A SX pump unavailable quantification.

<u>Compensatory Action Summary from the FPIE PRA Evaluation</u>

The following compensatory actions have been identified through review of the FPIE PRA results and are summarized below:

- Perform Operating Crew briefings on the actions to restore main feedwater and throttle the SX007 valves as needed.
- Perform Operating Crew briefings on the action to trip the RCPs to preclude damage to the Shutdown Seals for the duration of the extended CT.
- Prohibit planned maintenance activities on the 1A, 1B, 2A, and 2B DGs for the duration of the extended CT.
- Prohibit planned maintenance activities on the SX 1A, 1B, and 2B pumps.
- Prohibit planned maintenance activities on the motor-driven startup Feedwater pump, 2FW02P, for the duration of the extended CT. Note that this action is not credited in the 2A SX pump unavailable quantification.
- Prohibit planned maintenance activities on the Unit 2 SATs for the duration of the extended CT. Note that this action is not credited in the 2A SX pump unavailable quantification.

3.2.2 Opposite Unit Impact

Due to the crosstie capability of SX, the proposed extended CT was also analyzed for impact on Unit 1. As expected, the Unit 2 results for ICCDP and ICLERP were limiting. The results for Unit 1 are shown below in Table 3.2-8. The results reported include all restricted maintenance terms as listed in Table 3.1-2.

Table 3.2-8
FPIE PRA Risk Assessment Output
Results For Unit 1

Risk Metric	Value
$ICCDP_{2A\ SX\ U1}$	1.2E-08
$ICLERP_{2A\ SX\ U1}$	4.0E-10

3.2.3 Peer Review Finding IFSO-A4-01 Sensitivity Analysis

A disposition of peer review F&Os was performed as part of the PRA Model Adequacy study in Section 4. This review determined that one Finding, IFSO-A4-01, had the potential to impact the extended CT results. IFSO-A4-01 reads as follows:

Effect of plant-specific maintenance practices on internal flooding: Though this may only be a documentation issue, in absence of supporting information for not considering maintenance-induced flooding, an increase in internal flood frequencies of approximately 1.45 could apply.

To ensure this Finding does not have a significant impact on the results, a sensitivity analysis was performed by increasing all internal flooding frequencies by a factor of 1.45 (with an adjustment of 0.95 to account for plant capacity factor). This adjustment was made in the PRA model database and renamed as BW011b4-FloodSens.rr. Using this adjusted database, the ICCDP and ICLERP calculations were re-performed for both the base case and the 2A SX extended CT configuration shown in Table 3.1-2. These results are shown in Tables 3.2-9 and 3.2-10.

Table 3.2-9
FPIE Risk Assessment Input Parameters and Results for IFSO-A4-01 Sensitivity Case

Input Parameter	Value
$CDF_{BASE, FL SENS}$	1.99E-05/yr(1)
$CDF_{2A SX, FL SENS}$	2.31E-05/yr(1)
$LERF_{BASE, FL SENS}$	9.71E-07/yr(2)
$LERF_{2A SX, FL SENS}$	1.02E-06/yr(2)
$CT_{NEW, FL SENS}$	2.28E-02 yrs

(1) Based on a truncation of 1E-10

(2) Based on a truncation of 1E-11

Table 3.2-10
FPIE PRA Risk Assessment Output Results for IFSO-A4-01 Sensitivity Case

Risk Metric	Value
$ICCDP_{FL-SENS}$	7.3E-08
$ICLERP_{FL-SENS}$	1.1E-09

The sensitivity results were then compared to the original results to determine significance of the impact. This is shown in Table 3.2-11

Table 3.2-11
Percent Change for IFSO-A4-01 Sensitivity Case

	2A SX Extended CT	Flooding Sensitivity	Percent Change
ICCDP	7.2E-08	7.3E-08	**1.4%**
ICLERP	1.1E-09	1.1E-09	**1.5%**

The sensitivity analysis results show a minimal change in ICCDP and ICLERP. Therefore, it is determined that this open F&O does not have a significant impact on the evaluation for extending the 2A SX CT.

3.3 EXTERNAL EVENTS

3.3.1 Assessment of Relevant Hazard Groups

The purpose of this portion of the assessment is to evaluate the spectrum of external event challenges to determine which external event hazards should be explicitly addressed as part of the Braidwood 2A SX CT extension risk assessment.

The at-power PRA models used for this analysis include:

- internal events and internal floods, and
- internal fires.

In addition, the seismic, high winds, external floods, and other hazard groups are addressed qualitatively. It is noted that it is unnecessary to evaluate the low-power and shutdown contribution to the base CDF and LERF since the change being proposed involves performance of the repair while at-power. Because a detailed low power and shutdown PRA model has not been developed for this plant, the analysis conservatively omits the risk reduction that would typically occur with the unit shutdown.

3.3.2 Internal Fires

The impact on the internal fire risk profile due to the proposed CT extension is evaluated using the Braidwood Interim FPRA [Ref. 10], BB011b-FL-B. The Braidwood FPRA is an interim implementation of NUREG/CR-6850 and other approved methodologies in that the model documentation has not been finalized. However, the model has undergone peer review by a review team assembled by the PWR Owners Group and changes as a result of Findings and Observations from the peer review impacting model quantification have been incorporated in the model. The final documentation for this model is expected to be issued by the end of 2016. Therefore, the FPRA is judged useful to develop both quantitative and qualitative insights for this risk assessment.

The same process in Section 3.2 that was used for the FPIE model has also been used with the FPRA model results. The basic event changes for the equipment configuration during the extended CT are as shown in Table 3.1-2 for the SX A outage. The relevant inputs to Equation 3-1 are shown in Table 3.3-1 below. The corresponding output parameters from the equation above are then provided in Table 3.3-2. Note that equations apply to fire LERF as well and the relevant inputs are also shown in Table 3.3-1 with the output parameters provided in Table 3.3-2.

The fire risk insights and compensatory measures are focused on CDF since the results indicate that the impact on CDF risk measures is more significant than that associated with the fire impact on LERF risk. The ICFCDP due to fire is larger than for internal events; however, there is still considerable margin (i.e., more than a factor of 3) to the acceptance guidelines of 1.0E-5 (with implementation of effective compensatory measures).

Table 3.3-1
FIRE Risk Assessment Input Parameters

Input Parameter	Value
$FCDF_{BASE}$	5.32E-5/yr(1)
$FCDF_{SX\ A}$	1.66E-4/yr(1)
$FLERF_{BASE}$	5.96E-6/yr(1)
$FLERF_{SX\ A}$	7.65E-6/yr(1)
CT_{NEW}	2.28E-02 yrs

(1) Based on a truncation of 1E-11 for CDF and 1E-12 for LERF

Table 3.3-2
FIRE PRA Risk Assessment Base Output Results

Risk Metric	Value
$ICFCDP_{2A\ SX}$	2.6E-06
$ICFLERP_{2A\ SX}$	3.9E-08

A review of cutsets and importance measures is also performed to help understand the FPRA results. The FPRA results for CDF with 2A SX OOS indicate that the top cutsets are associated with failure to trip the RCPs or the loss of RCP seal cooling.

Significant Fire Zones and Compensatory Measures

The fire CDF results from the 2A SX OOS case identified the fire zones that could result in an increased likelihood of core damage during the extended 2A SX outage window. The fire zones with a contribution of greater than or equal to 1% of the 2A SX OOS risk are listed in Table 3.3-3. These fire zones would potentially benefit from additional compensatory measures that could further reduce the risk of fires in these zones.

From a qualitative perspective, the high fire risk contributors for when 2A SX is unavailable are expected from fire scenarios that impact the opposite division of SX (2B SX). This is in line with the top fire zones listed in Table 3.3-3 where fire zones from Division 2 are shown to be most important during the 2A SX outage window.

Table 3.3-3
Fire CDF 2A SX OOS Significant Fire Zones

Fire Zone	Fire Zone Description	Importance Contribution
5.1-2	Division 22 ESF Switchgear Room	26%
5.1-1	Division 12 ESF Switchgear Room	10%
3.2-0	Auxiliary Building El. 439'-0"	9%

Table 3.3-3
Fire CDF 2A SX OOS Significant Fire Zones

Fire Zone	Fire Zone Description	Importance Contribution
11.4-0	Auxiliary Building General Area, El. 383'	6%
11.6-2	Division 22 Containment Electrical Penetration Area, El. 426'	5%
11.2C-2	Containment Spray Pump 2B Room	4%
11.1B-0	Unit 2 Auxiliary Building Basement El. 330'	4%
18.10D-2	Unit Auxiliary Transformer 241-2	1%
18.10E-2	System Auxiliary Transformers 242-1/242-2	1%

Heightened awareness in the form of shift briefs or pre-job walkdowns will be implemented to reduce and manage transient combustibles prior to entrance into the extended CT. Additionally, hot work will be limited in these areas during the extended 2A SX outage window. This heightened awareness when combined with the other compensatory actions will reduce the potential for core damage from postulated fire scenarios.

As part of the Braidwood Configuration Risk Management Program (CRMP), Risk Management Actions (RMAs) were identified to reduce the fire risk when equipment with an appreciable impact on core damage mitigation is taken out-of-service. The BW CRM Program includes RMAs for when 2A SX is taken OOS for longer than 48 hours and are documented in BW-CRM-115, Revision 1 [Ref. 11]. For fire zones with high contribution, as specified in Table 3.3-3 above, RMAs that will be performed include maintaining detection and suppression systems, minimizing transient combustibles, maintaining fire zone barriers, and prohibiting hot work and temporary heat sources, prohibiting maintenance activities on certain panels, and avoiding switching at certain panels (as applicable, to the maximum extent possible without jeopardizing plant safety).

Significant Operator Actions and Compensatory Measures

The fire CDF results from the 2A SX OOS case identified the operator actions, if failed, that could result in an increased likelihood of core damage during the extended 2A SX outage window. The top five (5) operator actions with the greatest contribution are listed in Table 3.3-4.

Table 3.3-4
Fire CDF 2A SX OOS Significant Operator Actions

Operator Action	Description	Contribution
COMBO236-2	2RC-PMTRIPAHSYOA-F (see independent action below for a more detailed description); 0SX-XTIE---HMVOA-F (see independent action below for more detailed description)	24%
2RC-PMTRIPAHSYOA-F	OPS FAIL TO TRIP RCP TO PROTECT SDS	18%
COMBO210-2	2AF-AF005--HAVOA-F (see independent action below for more detailed description); 0SX-XTIE---HMVOA-F (see independent action below for more detailed description)	15%
0SX-XTIE---HMVOA-F	OPERATORS FAIL TO OPEN SX UNIT XTIE VALVES	3%
2AF01PB-FO-HXVOA-F	OPERATORS FAIL TO REFILL DDAFP FUEL OIL DAY TANK FROM STORAGE TANK - FIRE	3%
COMBO127-2	2AF-AF005--HAVOA-F (see independent action above for a more detailed description); 0SX005-----HMVOA-F (OPERATOR ACTION TO OPERATORS FAIL TO RECOVER SX005 X-TIE MOVs UPON LOSS OF POWER)	3%
2AF-AF005--HAVOA-F	OPERATORS FAIL TO OPEN AF005 VALVES (LOCALLY FAIL AIR) - FIRE	2%
COMBO602-2	2RC-PMTRIPAHSYOA-F (see independent action above for a more detailed description); 2FP-PRI-7X-HMVOA-F (OPERATORS FAIL TO ALIGN FP SEAL COOLING - SX NON-PIPE FAILURE INITIATOR)	1%

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.

<u>Summary of Compensatory Measure Impacts on Important Fire Zones</u>

Based on a review of results from the fire PRA contributors, the following compensatory actions are highlighted as important to reduce the risk from fire events during the performance of the extended CT:

- 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 walkdowns will be conducted to reduce and manage transient combustibles and to alert the staff about the increased sensitivity to fires in the following fire zones during the extended SX 2A outage window. Operating crew shift briefings will continue to be conducted every shift throughout the duration of the CT period. Additionally, planned hot work activities in the following 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 walkdowns 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).

Fire Zone[(1)]	Fire Zone Description
5.1-2	Division 22 ESF Switchgear Room
5.1-1	Division 12 ESF Switchgear Room
3.2-0	Auxiliary Building El. 439'-0"
11.4-0	Auxiliary Building General Area, El. 383'
11.6-2	Division 22 Containment Electrical Penetration Area, El. 426'
11.2C-2	Containment Spray Pump 2B Room
11.1B-0	Unit 2 Auxiliary Building Basement El. 330'
18.10D-2	Unit Auxiliary Transformer 241-2
18.10E-2	System Auxiliary Transformers 242-1/242-2

(1) For larger fire zones walkdowns may be focused on specific fire sensitive areas within the larger firezones. Walkdowns are judged as not being required for areas with continuous operator occupation (e.g. MCR). Fire Risk Management Actions (RMAs) where they occur may address the need for walkdowns in some of these areas. ALARA principles apply when reviewing radiological areas such as RHR.

- Risk Management Actions (RMAs) 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).

The Fire PRA risk for the 2A SX OOS condition discussed in this section will be reduced below reported values through implementation of these additional controls.

Opposite Unit Impact

Due to the crosstie capability of SX, the proposed extended CT was also analyzed for impact on Unit 1. As expected, the Unit 2 results for ICCDP and ICLERP were limiting. The results for Unit 1 are shown below in Table 3.3-5. The results reported include all restricted maintenance terms as listed in Table 3.1-2.

Table 3.3-5
Fire PRA Risk Assessment Output Results for Unit 1

Risk Metric	Value
$ICFCDP_{2A\ SX\ U1}$	2.8E-08
$ICFLERP_{2A\ SX\ U1}$	1.5E-09

3.3.3 Seismic

There is no quantitative Seismic PRA Model of Record for Braidwood. A Phase 1 seismic PRA (SPRA) model using non-plant-specific SSC fragility information was completed in 2013. However, this model, using the representative SSC fragility information, is not intended for quantitative evaluations. Thus a qualitative assessment must be performed. The Phase 1 seismic PRA model risk insights documented in BB-PRA-021.021.01, Revision 0, "Level 1 Seismic Quantification Notebook" [Ref. 12] was reviewed and considered in this qualitative assessment.

A fundamental concept of the seismic modeling is that similar components at the same location of a building will experience similar seismic forces. For Braidwood, station, all four SX pumps are located on the basement elevation of the Auxiliary Building and would be expected to experience the same seismic acceleration from a seismic event. In addition, with each pump being the same style, make, and model, the pumps would be expected to have the same fragility. Therefore, any seismic event that would result in failure of one SX pump would be expected to result in the failure of all SX pumps. Based on this information, assuming that the SX pumps are seismically correlated is reasonable. For the purposes of a seismic risk evaluation, the SX pumps can be treated as completely seismically correlated (i.e., a seismic event that can cause failure of one SX pump is likely to cause failure of all SX pumps).

Seismic hazards are estimated to be negligible contributors in this risk evaluation for the 2A SX pump being unavailable and are not included in the quantified risk results. 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. Motor driven pumps generally have high seismic capacity and a relatively low probability of failure compared to other components during a seismic event.

Therefore, unavailability of the 2A SX pump would not have a significant impact on overall seismic risk. This assessment concludes that seismic risk can be appropriately screened as a non-significant contributor to the risk of the proposed CT extension.

3.3.4 High Winds

Similarly to the Seismic discussion above, Braidwood station does not have a peer reviewed high winds hazard PRA model. The impact of the proposed completion time extension will be addressed qualitatively for high winds hazards.

As noted in the Braidwood IPEEE report [Ref. 13] the Braidwood station design is such that risk from high winds and tornadoes is not significant based on the screening assessments performed in [Ref. 13] which illustrated compliance with NRC SRP requirements. This assessment concludes that high winds risk can be appropriately screened as a non-significant contributor to the risk of the proposed CT extension.

3.3.5 External Flood

From the Braidwood IPEEE [Ref. 13], external floods were screened out as an acceptably small risk. Additionally, significant external floods would be a slow developing event which would be expected to allow time for restoration of the out of service 2A SX pump prior to presenting a significant challenge. The redundant SX pump room (i.e.1B and 2B SX pumps) is equipped with dual watertight doors which provides further mitigation of internal and external flooding.

3.3.6 Other External Hazards and Conclusions

As noted in the IPEEE [Ref. 13], the risk impact from Transportation and Nearby Facilities is negligible and has been screened from further consideration as documented in the SER and UFSAR for Braidwood station.

In evaluating the risk impact associated with nearby facilities, [Ref. 13] identified that the NRC SER concluded that the facilities do not represent a significant risk to the plant.

Based on the conclusions documented in [Ref. 13], the risk impact from Transportation and Nearby Facilities is insignificant and can be screened from further consideration.

As discussed in [Ref. 13], external events characterized as "high winds, floods and other external events" in NUREG 1407 [Ref. 14] have been evaluated using the guidance provided in that document. Those evaluations have been made on the basis of the UFSAR and NRC's SER for Braidwood station. The external events discussed above have uniformly been found to be of negligible risk significance for Braidwood station. This assessment concludes that the risk from the external events assessed above can be appropriately screened as a non-significant contributor to the risk of the proposed CT extension.

3.4 RESULTS COMPARISON TO ACCEPTANCE GUIDELINES

Table 3.4-1 shows a comparison of the individual hazard group core damage risk metrics to the acceptance guidelines defined in Section 1.3.4.

Table 3.4-1
COMPARISON OF INDIVIDUAL HAZARD GROUP RESULTS TO ACCEPTANCE GUIDELINES

Figure of Merit	Value	Acceptance Guideline	Below Acceptance Guideline
Internal Events and Internal Floods			
ICCDP	7.2E-08	<1.0E-06, or <1.0E-5[(1)]	Yes
ICLERP	1.1E-09	<1.0E-07, or <1.0E-6[(2)]	Yes
Internal Fires			
ICCDP	2.6E-06	<1.0E-06, or <1.0E-5[(1)]	Yes[(1)]
ICLERP	3.9E-08	<1.0E-07, or <1.0E-6[(2)]	Yes
Other Hazard Groups			
ICCDP	Negligible	<1.0E-06, or <1.0E-5[(1)]	Yes
ICLERP	Negligible	<1.0E-07, or <1.0E-6[(2)]	Yes
Total Values			
ICCDP	2.7E-06	<1.0E-06, or <1.0E-5[(1)]	Yes[(1)]
ICLERP	4.0E-08	<1.0E-07, or <1.0E-6[(2)]	Yes

(1) Per RG 1.177 a value between 1E-06 and 1E-05 may be deemed acceptable with effective compensatory measures implemented to reduce the sources of increased risk.

(2) Per RG 1.177 a value between 1E-07 and 1E-06 may be deemed acceptable with effective compensatory measures implemented to reduce the sources of increased risk.

The results indicate that the acceptance guideline values for a one-time extension are not exceeded for the ICCDP and ICLERP risk metrics. The internal events ICCDP and ICLERP results are far below the threshold for the acceptance guidelines, while the total values for ICCDP and ICLERP fall within the acceptable range with effective compensatory measures. Additional compensatory measures would potentially reduce risk further, such as protected equipment and heightened awareness of important operator actions and high risk fire zones. These additional measures are not accounted for in the quantification.

3.5 UNCERTAINTY ASSESSMENT

This section evaluates epistemic uncertainties that could impact the 2A SX CT extension assessment. Epistemic uncertainty is generally categorized into three types — parameter, model, and completeness uncertainty. These are each discussed in the sections which follow.

3.5.1 Parametric Uncertainty

Consistent with the ASME/ANS PRA Standard, quantitative parametric uncertainty analyses for both CDF and LERF were evaluated to determine if the point estimates calculated by the BB011b PRA model appropriately represent the mean. The uncertainty analysis did not have any F&Os from the most recent peer review related to parametric uncertainty. The results of the parametric uncertainty analyses confirm that the point estimate is a sufficient representation of the mean to represent the mean for the calculation of the changes in the risk metrics for this application. Please note that the Braidwood models BB011a (periodic update), BB011b (interim update) and BB011b4 (ASM for RCP Shutdown Seals) all use the same plant-specific and generic data for the development of random failure probabilities and maintenance unavailabilities with the exception of the additions made to address the revised RCP Seal LOCA modeling. The parametric uncertainty conclusions are considered applicable to all three models.

The same conclusion is applicable to the Fire PRA model uncertainty which was evaluated in the peer review and demonstrated a small variation in uncertainty relative to the point estimate. This same conclusion is considered applicable to the SX pump out of service condition evaluated in this evaluation.

3.5.2 Model Uncertainty

An evaluation of model uncertainty also exists for the BB011b PRA model. The documentation of the BB011b4 application specific model [Ref. 7] adds to this evaluation of model uncertainty via identification of key modeling assumptions related to the Shutdown Seals. These two sources of model uncertainty were reviewed for impacts on this specific application.

From the BB011b model uncertainty evaluation, the assumed alignment of SX pumps was identified as a potential model uncertainty. Since this evaluation applies the actual SX pump alignments that will be in place for Unit 2, this model uncertainty is not expected to impact the evaluation.

From the BB011b4 application-specific model documentation, the key model uncertainty is the modeling of the Shutdown Seals in the PRA model. Because a loss of SX can impact RCP seal cooling and lead to a challenge to the RCP seals, the modeling of the Shutdown Seals and associated human actions is identified as a potentially key uncertainty for this application. The modeling follows the guidance in PWROG-14001-P, Revision 1 [Ref. 20], and PWROG-14006-P, Revision 0-B [Ref. 21]. In addition, the analysis incorporates the technical issues from Westinghouse Technical Bulletin TB-15-1 [Ref. 22], which addresses required actions to maintain the No. 2 RCP seal integrity following loss of all seal cooling scenarios. The logic model for the existing non-SDS seals is based on WCAP-16141 [Ref. 23], also known as the WOG-2000 model. Therefore, the seal modeling follows the latest industry guidance, so the results are evaluated to realistically represent the as-built, as-operated plant.

3.5.3 Completeness Uncertainty

Section 3.3 addresses those hazard groups not included in the FPIE PRA. With the exception of internal fire, the majority of those hazard groups were qualitatively determined to have negligible impact on plant risk for the 2A SX CT extension.

As discussed in Section 3.3, the Braidwood Fire PRA used in this 2A SX CT assessment is an interim implementation of NUREG/CR-6850 and other approved methodologies. The Braidwood FPRA is judged sufficiently complete to provide useful risk insights for applications, including this SX CT extension. The interim FPRA model was utilized to obtain quantitative risk metric results, but more importantly it helped to identify those fire areas that were subject to increased risk from fire during the extended CT for consideration of potential compensatory measures.

The FPRA model utilized for the assessment 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 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 based on a CFAST model of the Main Control Room Complex spaces. Specific consideration of hot gas layer (HGL) and Multi-Compartment Analysis (MCA) is also included in the FPRA.

The potential for multiple spurious operations is included in the fire model, based on an expert panel evaluation of the generic PWR MSO scenarios and consideration of plant specific scenarios.

Instrumentation has been explicitly included in the fire PRA.

Based on the above information, there is no major form of completeness uncertainty that is judged to change the results of this assessment (i.e., ICCDP and ICLERP).

3.5.4 Uncertainty Analysis Conclusions

As previously indicated, the uncertainty analysis addresses the three generally accepted forms of uncertainty - parameter, model, and completeness uncertainty. The conclusions from these assessments are as follows.

Parameter Uncertainty

The parameter uncertainty assessment indicated that the use of the point estimate results directly for this assessment is acceptable.

Model Uncertainty

The model uncertainty assessment highlighted the following attributes as related to uncertainty being important to address with potential compensatory measures:

- The alignment of SX is fixed for this evolution, eliminating uncertainty for alignment of the SX system.
- The model utilizes the latest industry guidance for modeling of the RCP safe shutdown seals, so results are expected to adequately represent the plant as-built, as-operated.

Completeness Uncertainty

There is no major form of completeness uncertainty that would impact the results of this assessment. Although the fire model is an interim implementation of NUREG/CR-6850, it is essentially complete and appropriate for developing useful insights for the development of compensatory actions.

3.6 RISK SUMMARY

This analysis demonstrates with reasonable assurance that the proposed TS change is within the current risk acceptance in RG 1.177 for one-time changes. As shown in Table 3.4-1, there is significant margin between the calculated FPIE and FPRA risk metrics and the acceptance criteria considering the implementation of effective compensatory measures. The quantitative results combined with effective compensatory measures to maintain lower risk ensure the proposed TS change meets the intent of the ICCDP and ICLERP acceptance guidelines.

4.0 TECHNICAL ADEQUACY OF PRA MODEL

The 2015 application specific PRA model (BB011b4) is the most recent evaluation of the risk profile at BW for FPIE challenges. This model was developed to incorporate the installation of RCP safe shutdown seals at Braidwood. The BW PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the BW PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the BW PRA.

4.1 PRA QUALITY OVERVIEW

The quality of the BW FPIE PRA is important in making risk-informed decisions. The importance of the PRA quality derives from NRC Policy Statements as implemented by RGs 1.174 and 1.177, rule-making and oversight processes. These can be briefly summarized as follows using the words of the NRC Policy Statement (1995):

1. *"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art...and supports the NRC's traditional defense-in-depth philosophy."*

2. *"PRA...should be used in regulatory matters...to reduce unnecessary conservatism..."*

3. *"PRA evaluations in support of regulatory decisions should be...realistic...and appropriate supporting data should be publicly available for reviews."*

4. *"The Commission's safety goals...and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments..."*

5. *"Implementation of the [PRA] policy statement will improve the regulatory process in three ways:*

   - *Foremost, through safety decision making enhanced by the use of PRA insights;*

   - *Through more efficient use of agency resources; and*

   - *Through a reduction in unnecessary burdens on licensees."*

PRA quality is an essential aspect of risk-informed regulatory decision making. In this context, PRA quality can be interpreted to have five essential elements:

- <u>Scope (Section 4.2):</u> The scope (i.e., completeness) of the FPIE PRA. The scope is interpreted to address the following aspects:
  - Challenges to plant operation (Initiating Events):
    - Internal Events (including Internal Floods)
    - External Hazards
    - Fires
  - Plant Operational states:
    - Full Power
    - Low Power
    - Shutdown

- The metrics used in the quantification:
  - ➢ Level 1 PRA – CDF
  - ➢ Level 2 PRA – LERF
  - ➢ Level 3 PRA – Health Effects

- Fidelity (Section 4.3): The fidelity of the PRA to the as-built, as-operated plant.

- Standards (Section 4.4): ASME/ANS PRA Standard [Ref. 4 and Ref 5] as endorsed by the NRC in Regulatory Guide 1.200 [Ref. 1].

- Peer Review (Section 4.5): An independent PRA peer review provides a method to examine the PRA process by a group of experts. In some cases, a PRA self-assessment using the available PRA Standards endorsed by the NRC can be used to replace or supplement this peer review.

- Appropriate Quality (Section 4.6): The quality of the PRA needs to be commensurate with its application. In other words, the needed quality is defined by the application requirements.

4.2 SCOPE

The BW PRA is a full power, internal events (FPIE) PRA that addresses both CDF and LERF. The quantitative insights from the FPIE PRA are directly applicable to the 2A SX CT Extension PRA application. This scope is judged to be adequate to support the 2A SX CT PRA application. Consideration of other modes of operation is addressed in Section 1.4 and an evaluation of other potential hazard groups is included in Section 3.3

Because not all PRA standards are available to define the appropriate elements of PRA quality for all applications, the NRC has adopted a phased implementation approach. This phased approach uses available PRA tools and their quantitative results where standards are available and endorsed by the NRC. Where standards are not yet

available or endorsed, this approach uses qualitative insights or bounding approaches as needed.

The quality assessment performed in this section confirms the adequacy of the FPIE PRA. This quality assessment does not address the risk implications associated with low power or shutdown operation, nor does it address the quality assessment of external events. However, the results of the analysis for these other contributors have been used to obtain additional insights for potential compensatory measures and otherwise do not change the conclusions of the assessment.

Completion of a Fire PRA peer review within the past year and resolution of F&Os impacting risk quantification ensure the quality of the Fire PRA model used in this assessment. Completion of model documentation is expected by the end of this year.

4.3 FIDELITY: PRA MAINTENANCE AND UPDATE

The Exelon 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 plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management" [Ref. 15]) and subordinate implementation procedures. Exelon procedure ER-AA-600-1015, "FPIE PRA Model Update" [Ref. 16] delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites. The overall Exelon Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, Exelon risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on a four year cycle; shorter intervals may be required if plant changes, procedure enhancements, or model changes result in significant risk metric changes.

4.4 STANDARDS

The ASME/ANS PRA Standard provides the basis for assessing the adequacy of the Braidwood PRA as endorsed by the NRC in RG 1.200, Revision 2. The predecessor to the ASME/ANS PRA Standard was NEI 00-02 which identified the critical internal events PRA elements and their attributes necessary for a quality PRA.

4.5 PEER REVIEW AND PRA SELF-ASSESSMENT

There are three principal ways of incorporating the necessary quality into the PRA in addition to the maintenance and update process. These are the following:

- A thorough and detailed investigation of open issues and the implementation of their resolution in the PRA.
- A PRA Peer Review to allow independent reviewers from outside to examine the model and documentation. The ASME/ANS PRA Standard specifies that a PRA Peer Review be performed on the PRA.
- The use of the ASME/ANS PRA Standard to define the criteria to be used in establishing the quality of individual PRA elements.

There have been several assessments to support a conclusion that the Braidwood PRA model adequately meets the PRA standard such that it can be used to support risk applications in accordance with Regulatory Guide (RG) 1.200 Revision 2.

The Braidwood PRA model for internal events received a formal industry peer review in July 2013 [Ref. 9] against Addendum A of the ASME/ANS PRA Standard.

In the SE that was issued on February 24, 2011 from the NRC [Ref. 24] for implementation of the surveillance frequency control program (SFCP), which allows for relocation of surveillance test intervals to a licensee-controlled program, the following concluding statement was included regarding the quality of the Braidwood PRA model:

Based on the licensee's assessment using the applicable PRA standard and RG 1.200, the level of PRA quality, combined with the proposed evaluation and disposition of gaps, is sufficient to support the evaluation of changes proposed to surveillance frequencies within the SFCP, and is consistent with Regulatory Position 2.3.1 of RG 1.177.

It should be noted that PRAs can be used in applications despite not meeting all of the Supporting Requirements of the ASME/ANS PRA Standard. This is well recognized by the NRC and is explicitly stated in the ASME/ANS PRA Standard.

4.6 APPROPRIATE PRA QUALITY

The PRA is used within its limitations to augment the deterministic criteria for plant operation. This is confirmed by the PRA Peer Review and the PRA Self-Assessment. As indicated previously, RG 1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, consistency with applicable PRA Standards, relevant peer review findings, and the identification of key assumptions) is discussed below.

4.6.1 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE) is Exelon's PRA model update tracking database. These UREs are created for all issues that are identified with a potential to impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model. A review of the current open items in the URE database associated with plant changes for Braidwood is summarized in Table 4-1 along with an assessment of the impact for this application.

The results of the assessment documented in Table 4-1 are that none of the plant changes have any measurable impact on the SX CT extension request.

Table 4-1
Impact on the Braidwood PRA Model of Plant Changes Since the Last Model Update

URE Number	Description	Impact on the Application?	Disposition
BB-0230	ATWS Model revision in support of Power Uprate Project	No	ATWS is insignificant contributor to this application
BB-0815	Measurement Uncertainty Recapture (MUR) Power Uprate - ECs 378382 & 378383 (BYR), 378380 & 378381 (Bwd)	No	MUR represents an insignificant impact to results
BB-0903	Change in Steps in 1(2)BwEP-3, Steam Generator Tube Rupture	No	No impact on HRA for this application
BB-0963	HELB Removal of VX dampers and implement HELB dampers EC 388397 & EC 388442	No	Current model is conservative
BB-0986	Byron/Braidwood VA Recirculation Duct added to AUX BLD BY ECs 391071(2), BW ECs 388703,389632	No	No impact on this application
BB-1002	New Revision of 1(2)BWOA ELEC-3 Rev 102, analyze for HRA impact	No	No impact on HRA for this application.
BB-1006	CC pump discharge check valves replaced. The current valve is a Velon swing check with welded ends, while the new valve is a Crane nozzle check with flanged ends. The PRA uses plant-specific data for these check valves, which will no longer apply once the valves are replaced.	No	Negligible impact to this application
BB-1012	Annunciator window added in the control room which provides alarm for the VA monitoring system.	No	No impact to this application, model is conservative
BB-1014	ALTERNATE SX SUPPLY TO 1/2SX04P PUMP SUCTION	No	Model is conservative

Table 4-1
Impact on the Braidwood PRA Model of Plant Changes Since the Last Model Update

URE Number	Description	Impact on the Application?	Disposition
BB-1083	Review new subtask in BWOP RH-6 for HRA effect, BWOP RH-6 has a new subtask added that alters the HEP for 1RH-NR-SGTRHSYOA, OPERATORS FAIL TO ESTABLISH NORMAL RH SHUTDOWN COOLING	No	Current HEP value is conservative
BB-1084	Review new step in BWEP ES-1.3 for HRA effect. Step 12 was added to Rev 202 of ES-1.3 to specifically direct the operators to verify that CLR has been established. The HEP calculation associated with CLR alignment (1SI-HPR----HSYOA) currently uses a generic execution recovery as the verification procedure step did not exist at the time the calculation was developed. Applying step 12 as an execution recovery would lower the HEP to the mid E-3 range	No	Current HEP value is conservative
BB-1087	Modifications for FLEX	No	No impact to FPIE or FPRA, results are conservative
BB-1094	Multiple EC - MCR Fire Modifications, modifications are being made to respond to the NRC discovery of a circuit design deficiency for PORV response to a design basis MCR fire.	No	No impact to FPIE, FPRA is conservative
BB-1097	Relief valve along CV charging line introduces potential flow diversion during injection or as a LOCA pathway after downstream check valve failure.	No	Maximum flow diversion is 20 gpm, sensitivity run in MAAP5 showing negligible impact to success criteria for ECCS injection

Table 4-1
Impact on the Braidwood PRA Model of Plant Changes Since the Last Model Update

URE Number	Description	Impact on the Application?	Disposition
BB-1099	EH reservoir control block assembly remove and replace	No	No impact to FPIE, modification has not yet been installed (A2R19)
BB-1100	Fire mitigating actions to remove PORV control power fuses	No	No impact to FPIE, FPRA is conservative

4.6.2 Consistency with Applicable PRA Standards

As indicated above, a formal peer review against Addendum A of the ASME/ANS PRA Standard was performed in July 2013. The results of that review lead to the identification of the Braidwood PRA as not meeting Capability Category II for a small number of Supporting Requirements (SRs) listed below. These SRs are summarized in Table 4-2 along with an evaluation of their impact on the base model and this application.

The FPIE PRA model of record for this evaluation is Revision BB011b4 as documented in BB-ASM-002, Application Specific Model Notebook RCP Shutdown Seals [Ref. 7]. This application-specific model to incorporate the RCP Shutdown Seals is based on BB-PRA-014, Quantification Notebook, Revision BB011b [Ref. 8]. A peer review of the BB011b model was performed in July 2013 to assess the technical adequacy of the internal events and internal flooding models. The Peer Review report is documented in LTR-RAM-II-13-067-NP [Ref. 9].

LTR-RAM-II-13-067-NP identified six supporting requirements that were evaluated as not being met. In addition there were 10 supporting requirements that were assessed as being at Capability Category I. Table 4-2 provides a listing of the Not Met and Category I supporting requirements (SR) and an assessment of the impact on the evaluation presented here.

Table 4-2:
Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement	Capability Category	Evaluation Impact
DA-C5	Not Met	These SRs are associated with the counting of failures and demands in the development of failure probabilities. Changes in random failure probabilities are not expected to impact the calculation of the change in risk due to the component unavailability in this application.

Table 4-2: Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement	Capability Category	Evaluation Impact
DA-C6	Not Met	These SRs are associated with the counting of failures and demands in the development of failure probabilities. Changes in random failure probabilities are not expected to impact the calculation of the change in risk due to the component unavailability in this application.
IE-A7	Not Met	This SR deals with not including other than at-power events in the development of Initiating Events. A subsequent review has confirmed that no initiating events are missing. This SR does not impact the results of the evaluation.
IFQU-A6	Not Met	The assessment of this SR identifies that certain operator action timing information and the potential impact on cues was not addressed in the flooding assessment. The operator actions associated with the components addressed by this assessment are not impacted by the internal flooding actions.
IFSN-B3	Not Met	The assessment of this SR is related to the need for additional discussion of the inputs into the flood scenario development. Lack of this additional discussion does not impact the results provided in this evaluation.
LE-G5	Not Met	This assessment for this SR relates to a need for increased discussion of limitations in the LERF analysis that could impact applications. There are no specific limitations in the LERF analysis that impact this application.
HR-E3	CC I	This SR relates to having a structured review with Operations and training personnel to confirm the interpretation of the procedure is consistent with training expectations and plant use. This is not expected to have an impact on the results of this evaluation.
HR-E4	CC I	This SR relates to having a structured review with Operations and training personnel to confirm the interpretation of the procedure is consistent with training expectations and plant use. This is not expected to have an impact on the results of this evaluation.
IE-A8	CC I	This SR addresses interviews of plant personnel to determine if potential initiating events are missing from the PRA. The components being evaluated here do not have an impact on the modeled Initiating Events. A subsequent review has confirmed that no initiating events are missing. This SR does not impact the results of the evaluation.
IE-A9	CC I	This SR addresses review of plant-specific precursors to determine if potential initiating events are missing from the PRA. The components being evaluated here do not have an impact on the modeled Initiating Events. A subsequent review has confirmed that no initiating events are missing. This SR does not impact the results of the evaluation.
IFEV-A6	CC I	The assessment of this SR relates to a lack of inclusion of Braidwood specific OE in the development of internal flooding frequencies. The inclusion of this OE into the internal flooding results is not expected to have a significant impact on the model results or the results for this evaluation.

Table 4-2:
Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement	Capability Category	Evaluation Impact
IFSN-A6	CC I	The assessment of this SR relates to a lack of inclusion of HELB contributions to internal flooding. The inclusion of qualitative HELB considerations to the internal flooding results is not expected to have a significant impact on the model results or the results for this evaluation.
LE-F1	CC I	These SRs relate to the identification and documentation of significant contributors to LERF. The inclusion of this documentation does not impact the ability of the LERF model to provide accurate results. This SR does not impact the evaluation results.
LE-G3	CC I	These SRs relate to the identification and documentation of significant contributors to LERF. The inclusion of this documentation does not impact the ability of the LERF model to provide accurate results. This SR does not impact the evaluation results.
QU-D4	CC I	This SR relates to providing a comparison of quantification results to those from other plants. Since this assessment provides an evaluation on the delta risk impact, this SR will not have an impact on the evaluation results.
SC-A5	CC I	This SR relates to the assessment and development of beyond 24-hour mission times. The inclusion of these beyond 24-hour mission times is not expected to have a significant impact on the PRA model or on the results from this evaluation.

In summary, there were 16 Supporting Requirements that were judged to be "Not Met" or only meeting Capability Category I. As indicated in Table 4-2, these gaps have very limited or no impact on the model results and also have very limited or no impact on the SX CT extension request.

4.6.3 Relevant Peer Review Findings

RG 1.200, Revision 2 provides the following guidance with respect to meeting the ASME/ANS PRA Standard requirements and hence to the quality of a PRA model:

> If the requirement has been met for the majority of the systems or parameter estimates, and the few examples can be put down to mistakes or oversights, the requirement would be considered to be met. If, however, there is a systematic failure to address the requirement (e.g. component boundaries have not been defined anywhere), then the requirement has not been complied with. In either case, the examples of

noncompliance are to be (1) rectified or demonstrated not to be relevant to the application, and (2) documented.

The results of the July 2013 peer review are also used to identify the relevant peer review findings for the PRA model used for this assessment. These findings are summarized in Table 4-3 along with an assessment of the impact for the base model development. The associated F&Os with the "Not Met" or "Capability Category I" issues identified above are not repeated in this assessment. Table 4-3, therefore, only includes those "Findings" that are associated with SRs that were otherwise assigned to be at least Capability Category II from the peer review consistent with the RG-1.200 guidance quoted above.

Table 4-3:
Byron / Braidwood Finding F&Os from CC-II or Better Supporting Requirements

Finding F&O	Evaluation Impact
IE-A5-01	The SR requires that each system be analyzed systematically. There are systems or trains noted as screened and not causing a Plant Trip where there is not a detailed explanation or justification for no plant trip occurring. A subsequent review has confirmed that no initiating events are missing. This Finding does not impact the results of the evaluation.
IE-C1-01	Initiating Event SLOCA frequency does not include RCP Seal LOCA Frequency input. SLOCA frequency was re-examined to include random RCP seal failures in the current model used for this application.
AS-B3-01	Potential failure of containment sump suction screens due to debris clogging is not represented in the fault tree. This is not expected to be a significant effect since dominant sequences often already have failure of recirculation for other reasons.
SC-B1-01	Additional justification based on analysis other than MAAP is needed to justify large LOCA success criterion for RCS inventory control early. The Large LOCA success criteria is acceptable, but needs additional justification. Therefore, this Finding does not impact the results of the evaluation.
SC-B2-01	For loss of CC or SX, assumption of 30 minutes to trip RCP is not justifiable. Along with the new RCP Shutdown Seal modeling, the timing for operator actions has been updated within the current model used for this application.
SY-A15-01	The failure probability of SI8806 due to failure to remain open may rise above the 1% threshold per SY-A15 or even dominate other failure modes in the system. Quantification of the system gate 2SI-PUMPS-HPI has a value of 9.6E-4, while the probability of SI8806 spuriously closing would be 4.45E-08 * 24 hours = 1.07E-6, which is much less than the 1% threshold, so the impact of this Finding would be minimal.
SY-B12-01	HVAC dependency is not included for the switchgear and battery rooms. Modeling of the necessary dependencies is included in the current model.
SY-C2-01	Inconsistent references to battery life. This is a documentation issue only.
QU-D1-01	Documentation of cutset reviews. This is a documentation issue only.

Table 4-3:
Byron / Braidwood Finding F&Os from CC-II or Better Supporting Requirements

Finding F&O	Evaluation Impact
HR-A1-01	Review of Braidwood procedures for pre-initiator HRA in addition to Byron. Review of Braidwood-specific procedures showed no differences that would affect the PRA model, so this Finding has no impact on this evaluation.
HR-G4-01	It was found that there is only one bleed and feed cooling execution HEP developed, which uses thermal/hydraulic analysis timing based on non-loss of main feed water initiators. During an evaluation of the impact of this Finding, the highest cutset containing Loss of MFW and the HEP is in the form of one of its JHEPs (JHEP64) and occurs in the 1E-8 range at BW. Therefore, slight increases in the HEP or related JHEPs are not expected to have a significant effect on the model or this application.
DA-C3-01	Testing events for several systems are stated to satisfy the requirements for inclusion of testing in unavailability of the systems, yet the data are then stated to rely on values from the IPE study and not updated. Since the peer review, data was reviewed and sources were identified for most test & maintenance basic events in the current model. Inaccurate information was removed from Data Notebook regarding IPE as a source of current data.
DA-C3-02	Additional justification for and documentation of the final data used for the failure or unavailability data calculations is needed. This is a documentation issue only.
DA-C3-04	The use of plant-specific data from Byron for Braidwood or from Braidwood for Byron does not meet the requirement for the use of plant-specific data and should be avoided. Since any differences among these plants is minimal among those events using opposite-site data, the impact of this Finding is minimal.
DA-D7-01	SR-DA-D7 requires that if common cause events deleted from common cause population of estimate formula due to non-applicability events in the total population also have to be screened and deleted if non-applicable. It was noted that one common cause event excluded from the common cause group was not excluded from the total population. Changes in the one CCF term identified by the review team do not have a large overall effect on model results.
DA-E2-01	Document the processes used for data parameter definition, grouping, and collection. This is a documentation issue only.
IFSO-A1-01	Improve identification of flood sources. This is a documentation issue only.
IFSO-A1-02	Flood source from within SX valve pit. An analysis was performed to show that the bolts securing the deck plates for the SX valve pits can indeed withstand the hydrostatic pressure of water from the lake. In addition, a quantitative what-if analysis shows that the impact is expected to be in the E-8 range since it is only a few feet of piping. The impact of this Finding is minimal.
IFSO-A3-01	Review screening analysis for internal flooding. The screening criterion for miscellaneous buildings outside the power block was corrected to be associated with Supporting Requirement IFSN-A12 in Table B.2-1. This is a documentation issue only.
IFSO-A4-01	Effect of plant-specific maintenance practices on internal flooding. Though this may only be a documentation issue, in absence of supporting information for not considering maintenance-induced flooding, an increase in internal flood frequencies of approximately 1.45 could apply. This Finding was determined to have a potential impact on both the base model and the SX CT model. A sensitivity analysis was performed in Section 3.2.3

Table 4-3:
Byron / Braidwood Finding F&Os from CC-II or Better Supporting Requirements

Finding F&O	Evaluation Impact
IFEV-A5-01	Correct flooding initiating events for critical factor. Without the plant criticality factor, the results for each calculation are slightly conservative, and therefore have a minor impact on the model. This may work to offset some of the impact from IFSO-A4-01.
IFEV-A5-02	Correct inconsistent flood IE frequencies to consistently account for piping at a data threshold such as 6". The impact of these corrections is minor, and shifts frequencies to less severe scenarios, so the impact of this Finding is minor.
IFQU-A9-01	Address indirect effects of flooding, such as jet impingement. A subsequent analysis provides a discussion of the jet impingement forces imposed on insulated pipes surrounded by lagging due to a pipe rupture. The analysis concluded that the impingement forces for moderate energy water sources are insufficient to perforate the aluminum lagging and threaten damage to equipment located at a distance from the water source. Therefore, this Finding has no impact on the application.
IFQU-B1-01	Documentation for the Braidwood Station is not provided in the Internal Flooding Notebook. This is a documentation issue only.

In summary, there were 24 additional peer review findings not already encompassed within the entries in Table 4-2. As indicated in Table 4-3, however, most of these remaining open items have no or very limited impact on the model results and also should have no or very limited impact on most applications of the model. The one issue related to Internal Flooding frequencies (IFSO-A4-01) is addressed with a simple sensitivity assessment performed in Section 3.2.3.

4.6.4 Identification of Key Assumptions

Key assumption identification is addressed by the Uncertainty Assessment in Section 3.5, specifically the Model Uncertainty in Section 3.5.2 and Completeness Uncertainty in Section 3.5.3. The key assumptions that introduce uncertainties for this application are summarized in Section 3.5.4.

4.6.5 Fire PRA Peer Review Results and F&Os

The FPRA peer review for Braidwood was performed by the PWR Owners Group in May 2015. The results of this peer review are summarized in Tables 4-4 (Not Met and Capability Category I Supporting Requirements) and 4-5 (Finding F&Os from CC-II or Better Supporting Requirements) below. As noted in the evaluation of the supporting requirements and F&Os below, the issues identified have been addressed in the current quantification to the extent that they impact the quantification results.

Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
CS-B1	16-4	Based on information provided there is lack of details to meet capability category II/III as the only statement is in Section 3.8 of the 'Braidwood Fire PRA Cable Selection Notebook (BW-PRA-021.03), Rev 0' : "The BW Fire PRA reviewed the electrical coordination calculations for applicability to the Fire PRA. These were reviewed for each of the credited power supplies in the model." This did not provide Analysis or Identified any additional requirements only stated that it was reviewed. (This F&O originated from SR CS-B1)	FALSE	APPENDIX C OF THE BW CABLE SELECTION NOTEBOOK (BW-PRA-021.03) WAS UPDATED TO PROVIDE SPECIFIC REFERENCES AND ADDITIONAL DETAILS PERTAINING TO THE BREAKER COORDINATION OF THE CREDITED POWER SUPPLIES. THE INCLUSION OF THIS ADDITIONAL INFORMATION DID NOT CHANGE THE RESULTS OF THE BREAKER COORDINATION.
FQ-E1	19-9	HLR-QU-D7 requires review of importance of components and basic events to determine that they make logical sense. (This F&O originated from SR FQ-E1)	FALSE	Documentation upgrade only. No impact on quantification.

Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
FQ-E1	19-8	Document the relative contribution of contributors to LERF. (This F&O originated from SR FQ-F1) ***REMOVE THIS LINK TO SR FQ-E1***	FALSE	Documentation upgrade only. No impact on quantification.
FQ-E1	19-17	Nonsignificant accident cutsets (This F&O originated from SR FQ-E1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-E1	19-16	Perform a quantification evaluation for the contribution of contributors to LERF for the FPRA. (This F&O originated from SR FQ-E1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-E1	19-10	An internal event HFE is being used in the fire PRA in addition with the fire event HFE version. (This F&O originated from SR FQ-E1)	TRUE	See HRA F&Os. Updated HFEs as required.
FQ-F1	19-14	Limitations on knowledge of severe accident phenomenology as well as level-2 PRA modeling to capture severe accident progression has not been provided. (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-F1	19-1	There is no discussion of asymmetries from QU-F2(I). (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.

Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
FQ-F1	19-11	There is no document of the importance measures for Braidwood Unit 2 CDF/LERF from QU-F2(j). (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-F1	19-13	The definition of significance is not being used. There is no discussion of documentation of significance definitions (basic event, cutsets, and accident sequences) as required by QU-F6. (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-F1	19-15	Document the process used to identify plant damage states and accident progression contributors. (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FQ-F1	19-12	There is no discussion of the quantification process limitations as required in QU-F5. (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FSS-D7	14-6	Probabilities of suppression unavailability are credited, but not plant-specific information. (This F&O originated from SR FSS-D7)	FALSE	Data used is based on NUREG/CR-6850. A review of plant data confirmed no outlier behavior.
HRA	15-6	The Human Reliability Analysis notebook Appendix A identifies the cues (instrumentation and other indications) required for significant operator actions used in the Fire PRA. The instrumentation required is not consistent with the indicators in Appendix A Table A-1 of the Equipment Selection notebook.	FALSE	Instrumentation are explicitly modeled in the Byron and Braidwood fire model. Table 3-4 in the HRA notebooks summarizes the instrumentation for operator actions in the Fire PRA model.

**Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements**

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
HRA-A2	18-2	Plant fire response procedures were not reviewed to identify possible new fire-specific safe shutdown actions for the FPRA, including MCR abandonment. (This F&O originated from SR HRA-A2)	FALSE	Braidwood fire procedures have been reviewed and documented in Section 2.2.3 of the HRA Notebooks
HRA-A4	18-4	Errors of commission, due to a single spurious fire-induced failure, were not reviewed with Braidwood operators. The procedures and sequence of events were not reviewed in detail with Braidwood operations and training personel to confirm that the interpretation of the procedures was consistent with plant observations and training procedures. (This F&O originated from SR HRA-A4)	FALSE	Operator interviews have been performed at Braidwood and scenarios have been discussed with operators in detail. The outcome of the operator interviews is presented in Appendix D of the HRA Notebook.
HRA-E1	18-7	With respect to documenting the FPRA, the Fire PRA HRA Notebook (BW-PRA-021.09) contains inconsistencies and discrepancies, including: Only Unit 1 is documented, although some differences between the two units exist in the FPRA database. Some combination HEPs in Table 4.1 are different from those in the FPRA. Some documented screening HEPs are not in the FPRA. (This F&O originated from SR HRA-E1)	FALSE	Documentation, HRAC HEP values, and CAFTA model HEP values have been reviewed and inconsistencies have been resolved. Correct values are used in the CAFTA model.

Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
IGN-A4	20-4	Plant-specific fire events were not compared against the fire events used to develop the generic ignition frequencies in NUREG-2169. (This F&O originated from SR IGN-A4)	TRUE	No outlier events were identified, therefore, the use of NUREG-2169 is considered appropriate.
PRM-B11	15-14	The instrumentation and indications required for cues identified in the HRA notebook are not included in the Fire PRA model. (This F&O originated from SR PRM-B11)	TRUE	See HRA F&Os. All required cues are addressed in the FPRA model.
PRM-B15	15-15	BW-PRA-021.05, Fire PRA Plant Response Model notebook Rev 0, Section 3.1.8 and Appendix B. Appendix B documents the review of the containment paths and the basis for screening or including the individual pathways. Additional pathways were identified that should be included in the Fire PRA model. However, these pathways appear to not be included under gate 1(2)-CONT-ISOLATION. (This F&O originated from SR PRM-B15)	FALSE	The required logic was incorporated into the Fire model under the specified gate.
PRM-B2	15-9	Internal events F&O IE-A5-01 does not appear to have been addressed for the Fire PRA analysis. (This F&O originated from SR PRM-B2)	FALSE	Documentation upgrade only. No impact on quantification.
PRM-B2	15-12	Internal events F&O SC-A5-01 does not appear to have been addressed for the Fire PRA analysis. (This F&O originated from SR PRM-B2)	FALSE	Documentation upgrade only. No impact on quantification.

Table 4-4:
Braidwood Not Met and Capability Category I Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
PRM-B2	15-11	Internal events F&O HR-G4-01 does not appear to have been addressed for the Fire PRA analysis. (This F&O originated from SR PRM-B2)	FALSE	Documentation upgrade only. No impact on quantification.
PRM-B2	15-10	Internal events F&O SC-B2-01 does not appear to have been addressed for the Fire PRA analysis. (This F&O originated from SR PRM-B2)	FALSE	Documentation upgrade only. No impact on quantification.
UNC-A1	18-12	Some anomalies were observed in the database that was used for the uncertainty analysis which was different from that used in the FPRA. (This F&O originated from SR UNC-A1)	FALSE	Anomalies were addressed and do not impact conclusions of uncertainty analysis.
UNC-A2	18-13	Some documented assumptions were not identified as potential sources of uncertainty and characterized to identify the effect on the PRA model and to permit the potential impact on the results to be understood. (This F&O originated from SR UNC-A2)	FALSE	Documentation upgrade only. No impact on quantification.
UNC-A2	18-14	Based on similarities between the two plants, some elements of the Braidwood FPRA reflect the application of work performed for Byron, without identifying this as a potential source of uncertainty. (This F&O originated from SR UNC-A2)	FALSE	Documentation upgrade only. No impact on quantification.

Table 4-5: Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
CS-A1	16-1	Per review of Braidwood Fire PRA Detailed Circuit Analysis Notebook (BWPRA-021.03.01), Rev 0" and "Braidwood Fire PRA Cable Selection Notebook (BW-PRA-021.03), Rev 0" the captured cables did not include all required cables. (This F&O originated from SR C	TRUE	A REVIEW AGAINST THE REQUIRED CABLES IDENTIFIED IN BW(BY)-PRA-021.03.01 WAS PERFORMED. ANY CABLES THAT WERE MISSING FROM THE CABLE TO COMPONENT DATA WERE ADDED INTO THE FIRE PRA MODEL. THE CABLES THAT WERE IDENTIFIED TO BE MISSING SHOULD HAVE BEEN INCLUDE
ES-A4	15-2	Appendix A Table A-1 contains some dispositions, bases, and notes that are inconsistent, vague, or inaccurate. (This F&O originated from SR ES-A4)	TRUE	THE BE MAPPING FOR BOTH BY AND BW HAS BEEN UPDATED TO REFLECT THE NECESSARY CODING AND DISPOSITIONS CHANGES. THE NEW BE MAPPING CAN BE FOUND IN APP A OF THE EQUIPMENT SELECTION NOTEBOOK (BW(BY)-PRA-021.02.
ES-B1	15-4	In Appendix C Table C-1, component 0SX007, the disposition does not appear correct. (This F&O originated from SR ES-B1)	FALSE	Appendix C of the BW Equipment Selection notebook (BW-PRA-021.02) has been updated to reflect the modeling of this component. The disposition was changed to “modeled in PRA”.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
ES-B1	15-5	Appendix C Table C-1 contains several fire damper components contain notes that indicate that there may be open items for these components. (This F&O originated from SR ES-B1)	FALSE	THE DAMPER MAPPING HAS BEEN UPDTED TO REFLECT THE PRA MODELING. APP C OF THE EQUIPMENT SELECTION NOTEBOOK HAS BEEN UPDATED TO REFLECT THESE CHANGES (BW-PRA-021.02).
ES-D1	15-8	There are several incorrect references within the report that should be corrected. (This F&O originated from SR ES-D1)	FALSE	THE SECTION REFERENCES HAVE BEEN UPDATED TO REFLECT THE APPROPRIATE SECTION OR APPENDIX NUMBERS. THESE UPDATES WERE MADE IN BW-PRA-021.02.
FQ-B1	19-2	The use of modules in the fire risk quantification was not documented for interpretation. (This F&O originated from SR FQ-B1)	FALSE	INCLUDED STATEMENT IN SECTION 3.2 OF BW(BY)-PRA-021.11 CLARIFYING THAT MODULES ARE NOT USED IN THE FIRE PRA QUANTIFICATION.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
FQ-B1	19-3	The criteria of the truncation limit for CDF/LERF of being less than 5% for the final change is not met and there is a potential for accident sequences being inadvertently eliminated. (This F&O originated from SR FQ-B1)	TRUE	Updated quantification has confirmed convergence.
FQ-B1	19-4	Identfication and correction of mutually exclusive events in the results has not been defined. (This F&O originated from SR FQ-F1)	FALSE	Documentation upgrade only. No impact on quantification.
FSS-A1	14-3	When locating transient fires in PAUs, plant personnel indicated that transient scenarios were only placed in areas with targets such as conduits or cable trays. Transient fires were not placed in areas where the only targets were PRA equipment. The risk contribution from the areas excluded from transient ignition sources could possibly be significant. (This F&O originated from SR FSS-A1)	FALSE	A sensitivity analysis was performed placing transient ignition sources adjacent to fixed ignition sources (panels). The impact on risk was small using a conservative estimate of the impact.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
FSS-G3	14-4	The screening criteria was not applied correctly in one multi compartment scenario. (This F&O originated from SR FSS-G3)	TRUE	THE MCA INTERACTIONS HAVE BEEN REVIEWED TO IDENTIFY WHERE THE SCREENING CRITERIA WAS APPLIED INCORRECTLY. IN THOSE CASES THE INTERACTIONS WERE ADDED BACK TO THE LIST OF INTERACTIONS TO BE ANALYZED. THE UPDATED INTERACTION LIST IS PROVIDED IN APP A OF THE
FSS-H1	14-5	The methodology for assigning ignition frequency and targets for cable fires due to welding is not described in the scenario development notebook. (This F&O originated from SR FSS-H1)	FALSE	Documentation upgrade only. No impact on quantification.
FSS-H1	14-7	This is a generic F&O related to the overall documentation. (This F&O originated from SR FSS-H1)	FALSE	Documentation upgrade only. No impact on quantification.
HRA	15-6	The Human Reliability Analysis notebook Appendix A identifies the cues (instrumentation and other indications) required for significant operator actions used in the Fire PRA. The instrumentation required is not consistent with the indicators in Appendix A Table A-1 of the Equipment Selection notebook.	FALSE	Instrumentation are explicitly modeled in the Byron and Braidwood fire model. Table 3-4 in the HRA notebooks summarizes the instrumentation for operator actions in the Fire PRA model. Document clarification only.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
HRA-A1	18-1	Some FPIE actions were carried forward into the FPRA with no clear determination of relevance and validity within the context of the fire scenarios. (This F&O originated from SR HRA-A1)	TRUE	1RC-EB-ATWSHSYOA-F and 1RT-RX-ATWSHRBOA-F are set to 1.0 in the fire model. Internal events single HFEs and joint HFEs have been removed from the fire model and only fire versions are credited.
HRA-A1	18-3	The identification of the key human response actions was not always based on Braidwood-specific procedures. (This F&O originated from SR HRA-A1)	FALSE	Documentation upgrade only. No impact on quantification.
HRA-B1	18-9	The FPRA credits some operator actions which were not modified through screening or detailed analysis for applicable fire effects. (This F&O originated from SR HRA-B1)	TRUE	Risk significant HFEs set to screening values in the internal events model have been reviewed and specific fire HFEs have been developed. The assessment column in Table 5-1 of the Fire HRA Notebook identifies the new HFEs developed for the fire model.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
HRA-B3	18-11	There is no explicit listing of the particular instruments and indications that are credited to provide the required cues and no correlation of those cues to the instruments listed on the T-sheets as reliable for particular fires. (This F&O originated from SR HRA-B3)	FALSE	Table 3-4 of the HRA Notebook provides a list of instruments credited for each operator action.
HRA-B3	18-5	The development of some HFEs credited support systems that are inconsistent with assumptions for the FPRA. (This F&O originated from SR HRA-B3)	TRUE	1AF-START--HPMOA-F is based on the BB0003 run that has feed water unavailable. 1FW-FRH1---HSGOA-F assumed that MFW is not available. It discusses the possibility to start AFW to avoid CD after 55 min but this is not credited.
HRA-B3	18-6	The definitions of some specific high-level tasks do not describe cross-unit performance factors (e.g., availability of cues, communication, manpower) for credited train-level recoveries. (This F&O originated from SR HRA-B3)	FALSE	The fire HRA approach is consistent with the internal events HRA approach. Cross unit HFE applicability is described in FPIE HRA Notebook 5.2.1. Text at the end of paragraph 2.1.3 of the fire HRA notebook has been added to explain cross-unit impact.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
HRA-C1	18-10	Detailed analysis was not performed for some HFEs that were dispositioned in Table 5-1 as "Not risk significant in FPRA model" but for which the screening HEP met one or more criteria for being considered risk significant. (This F&O originated from SR HRA-C1)	TRUE	All risk significant HFEs have been revised and detailed analysis performed for HFE that had screening values in the internal events model.
HRA-C1	18-8	Some HFEs are included in the FPRA with no basis to establish the relevant fire-related effects for the associated HEPs. (This F&O originated from SR HRA-C1)	TRUE	All HFEs have been revised and HFEs that are credited in the model consider fire-related effects.
IGN-A7	20-1	During the Peer Review walkdown, three rooms were checked and found to have errors in ignition source counting: 11.4A-1, 11.4A-2, 11.6-0. (This F&O originated from SR IGN-A7)	TRUE	THE COMPONENT COUNT IN 11.4A-1 AND 11.4A-2 HAVE BEEN UPDATED TO THE CORRECT COUNTS BASED ON THE EQUIPMENT LOCATED IN THE PAU. THIS IS DOCUMENTED IN APPENDIX A OF BW-PRA-021.06 AND APPENDIX A OF BW-PRA-021.10. A REVIEW WAS ALSO COMPLETED ON OTHER IGNITION SOURCES TO IDENTIFY IF THIS ISSUE WAS FOUND ELSEWHERE. DISCREPANCEIS IDENTIFIED WERE CORRECTED AS NECESSARY. THIS REVIEW WAS DONE FOR BOTH BY AND BW.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
PP-B6	20-5	Figure 4.1 does not indicate any engineering manholes as PAUs. (This F&O originated from SR PP-B6)	TRUE	DISCUSSION ADDED IN SECTION 3.1 FOR MANHOLES, DRAWINGS ADDED AS APPENDIX C, OF NOTEBOOK PP (BY/BW)-PRA-021.01).
PP-C1	20-2	A review of Figure 4-1 of PP shows some possible buildings or structures that are not identified. (This F&O originated from SR PP-C1)	TRUE	ALL BUILDINGS HAVE BEEN IDENTIFIED WITHIN THE PLANT LAYOUT DRAWINGS. THIS INCLUDES ALL STRUCTURES THAT MAY BE RELATED TO TANKS OR ELECTRICAL TOWERS. SEE FIGURE 4-1, FIGURE 4-2, AND TABLE 4-1 OF THE BW PLANT PARTITIONING NOTEBOOK (BW-PRA-021.01).
PP-C3	20-3	Appendix B of the PP notebook gives the features of the partitioning elements, but it does not provide details of walkdowns. Plant personnel say they have the walkdowns documented on an IPAD. (This F&O originated from SR PP-C3)	FALSE	Documentation upgrade only. No impact on quantification.
PRM-B4	15-17	Logic for spurious opening of pressurizer PORVs (MSO-17) should be modified. Logic for spurious opening of steam generator relief valves (MSO-22) should be modified. (This F&O originated from SR PRM-B4)	TRUE	THE LOGIC FOR MSO17 HAS BEEN UPDATED TO REFLECT THAT A SINGLE PORV SPURIOUSLY OPENING CAN LEAD TO A SMALL LOCA. THIS CHANGE HAS BEEN DOCUMENTED IN APPENDIX A OF BW-PRA-021.05. SEE RESPONSE TO F&O PRM-B4 (21-7) FOR MSO22 CHANGES.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
PRM-B4	21-7	Logic for spurious opening of steam generator relief valves (MSO-22) should be modified. (Appendix A change #23G) (This F&O originated from SR PRM-B4)	TRUE	THE LOGIC MODEL HAS BEEN UPDATED TO ENSURE THAT ALL SLBO LOCATIONS INLCUDE THE MSO22 LOGIC AS AN INPUT. MSO22 WAS REMOVED FROM ALL SBLI LOCATIONS AS IT IS MORE APPROPRIATE TO MODEL THIS UNDER SBLO. THIS CHANGE IS DOCUMENTED IN APPENDIX A OF BY(BW)-PRA-021
PRM-C1	15-13	PRM notebook Section 3.1.5 states that no changes are made to the logic, but changes are made to the logic for fire-specific failures. (This F&O originated from SR PRM-C1)	FALSE	Documentation upgrade only. No impact on quantification.
QLS-A3	17-1	Section 3.3 of the Qualitative Screening Notebook (BW-PRA-021.04) describes the application of all the screening criteria to all PAUs. Appendix A lists the screening bases. Appendix B lists all PAUs from the Plant Partitioning Notebook (BW-PRA-021.01) and	FALSE	THE QUALITATIVE SCREENING CRITERIA HAS BEEN REVISED TO ADDRESS THE CONCERN OF SCREENING PAUS WITHOUT ADEQUATE CRITERIA MET. APPENDIX A AND APPENDIX B OF THE QUALITATIVE SCREENING NOTEBOOK (BW-PRA-021.04) HAVE BEEN UPDATED APPROPRIATELY.

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
SF-A1	17-2	In Section 4.1.1, various sections of the IPEEE are quoted skipping potential issues that were identified. A conclusion is made based entirely on the IPEEE work with no evaluations or discussions about the past potential issues with respect to the current FPRA work. (This F&O originated from SR SF-A1)	FALSE	Section 4.1.1 of BW(BY)-PRA-021.13 has been updated to include additional references that indicate the results and findings from seismic walkdowns.
SF-A2	17-3	Section 4.1.2 addressed fire water and carbon dioxide suppression systems but not Halon. IPEEE Section 4.9.3.1.2 was quoted as support for Spurious Actuation of a Fire Water System. (This F&O originated from SR SF-A2)	FALSE	Documentation upgrade only. No impact on quantification.
SF-A3	17-4	Section 4.1.3 describes the three suppression systems: water, carbon dioxide, and Halon. However, the potential for common cause failure is only described for water. In addition, a "not insignificant potential exists" for a common cause failure of the fire protection pumps due to loss of the Lake Screen House is mentioned in the FPRA. But there is no assessment of the risk to the plant. (This F&O originated from SR SF-A3)	FALSE	Section 4.1.3 of BW(BY)-PRA-021.13 has been updated to include a complete discussion of the potential for common cause failure of all suppression systems.
SF-A4	17-5	In Section 4.1.4, the concluding paragraphs reference the conclusions of Sections 4.1.1 and 4.1.2 but not 4.1.3 to justify the effectiveness of the plant seismic response procedures. (This F&O originated from SR SF-A4)	FALSE	Section 4.1.4 of BW-PRA-012.13 has been updated to reference section 4.1.3. This provides the discussion of common cause between the different systems.

**Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements**

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
SF-B1	17-6	The BW Seismic-Fire evaluation is documented in BW-PRA-021.13. Some documentation deficiencies have been identified in Sections 3.1 and 4.1. Section 3.1.1 doesn't discuss how seismic fire ignition sources were identified; all that is described is the Seismic PRA generally. Section 3.1.2 discusses how fire suppression was assessed but not fire detection. Section 3.1.3 incorrectly states that the fire suppression system is made up of a single fire water system. There are carbon dioxide and automatic Halon systems. Also, there is no discussion about how the evaluation was performed. Section 4.1.1 quotes various sections from the IPEEE with no explanation about why these specific sections are pertinent to seismic fire ignition source assessments. IPEEE Section 3.4.7.2 is the most applicable section from IPEEE. But the excerpt missed the discussion on seismically induced fires. Moreover, this excerpt is repeated in Sec. 4.1.1 with no discussion about the potential issues that were described in the IPEEE. Section 4.9.3.1 from the IPEEE is quoted with "Response" excerpts from sections 4.9.3.1.1, 4.9.3.1.2, and 4.9.3.1.3 (Seismically-Induced Fires, Seismic Actuation of Fire Suppression Systems, and Seismic Degradation of Fire Suppression Systems, respectively). It is unclear why fire suppression is discussed in this section. Also, the excerpts skipped the discussion of issues that were described in the IPEEE. It's unknown whether those issues are significant for this current Fire PRA assessment. In Section 4.1.2, the paragraph after the reference to IPEEE 4.9.3.1.2 in "Smoke Detector Actuation during a Seismic Event" is out of place; this paragraph belongs in the discussion on Spurious	FALSE	Revised Section 4 to add in discussions related to findings from the IPEEE. In most cases the direct quotes have been removed, and specific conclusions have been made based on the IPEEE.

Table 4-5: Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
		Actuation. In Section 4.1.4 there is the typo "liaison" that should be "liaise." Section 4.1.5 could be improved upon by adding more detail on the firefighting equipment, estimated response times, and any outside agency support if applicable. In addition, the opening paragraph in Section 4.1.5 could be rewritten since it seems out of place. (This F&O originated from SR SF-B1)		

Table 4-5:
Braidwood Finding F&Os from CC-II or Better Supporting Requirements

SR	F&O NO.	DESCRIPTION	QUANT IMPACT	ACTION TAKEN
CS-B1	16-4	Based on information provided there is lack of details to meet capability category II/III as the only statement is in Section 3.8 of the 'Braidwood Fire PRA Cable Selection Notebook (BW-PRA-021.03), Rev 0' : "The BW Fire PRA reviewed the electrical coordination calculations for applicability to the Fire PRA. These were reviewed for each of the credited power supplies in the model." This did not provide Analysis or Identified any additional requirements only stated that it was reviewed. (This F&O originated from SR CS-B1)	FALSE	APPENDIX C OF THE BW CABLE SELECTION NOTEBOOK (BW-PRA-021.03) WAS UPDATED TO PROVIDE SPECIFIC REFERENCES AND ADDITIONAL DETAILS PERTAINING TO THE BREAKER COORDINATION OF THE CREDITED POWER SUPPLIES. THE INCLUSION OF THIS ADDITIONAL INFORMATION DID NOT CHANGE THE RESULTS OF THE BREAKER COORDINATION.

4.7 GENERAL CONCLUSION REGARDING PRA CAPABILITY

The Braidwood PRA maintenance and update processes and technical capability evaluations provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions, specifically in support of the requested extended CT for the SX system.

Previously identified gaps to specific requirements in the ASME/ANS PRA Standard have been reviewed to determine which gaps might merit application-specific sensitivity studies in the presentation of the application results. One sensitivity study was performed for internal flooding initiators per Finding F&O IFSO-A4-01 (documented in Section 3.2.3) and was found to not have significant impact on this SX CT extension request.

5.0 SUMMARY AND CONCLUSIONS

5.1 SCOPE INVESTIGATED

This analysis evaluates the acceptability, from a risk perspective, of a change to the Braidwood TS for the 2A Essential Service Water Train (2A SX) for a one-time increase of the CT from 72 hours to 200 hours when 2A SX is inoperable.

The analysis examines a range of risk contributors as shown in Table 5-1.

Table 5-1
SUMMARY OF RISK INSIGHTS FOR 2A SX CT EXTENSION

RISK CONTRIBUTOR	APPROACH	INSIGHTS
Internal Events	Quantify ICCDP & ICLERP for planned configuration • ICCDP < 1E-6 • ICLERP < 1E-7 If exceeded compare to acceptance guidelines with risk management actions implemented to reduce sources of risk • ICCDP < 1E-5 • ICLERP < 1E-6	• Base risk well within acceptance guidelines • Compensatory measures further reduce risk
Internal Fire	Qualitatively and quantitatively evaluated: • Identify fire scenarios impacted by configuration • Estimate fire risk impacts due to configuration and quantify ICCDP and ICLERP • Identify compensatory measures	• ICCDP and ICLERP within acceptance guidelines with risk management actions to reduce risk sources. • Internal events compensatory measures apply to fire scenarios • Additional Fire-related compensatory measures identified
Seismic	Qualitatively evaluated.	• Seismic risk impacts negligible

Table 5-1
SUMMARY OF RISK INSIGHTS FOR 2A SX CT EXTENSION

RISK CONTRIBUTOR	APPROACH	INSIGHTS
High Winds	Qualitatively evaluated.	• High winds risk impacts negligible • High winds risk reduced with compensatory measures for internal events and fire
Other External Hazards	Qualitatively evaluate each hazard based on the BW IPEEE and a re-examination for the specific configuration with 2A SX inoperable.	• Other External Event risks were found to be negligible contributors
Overall At-Power Risks	Quantify ICCDP & ICLERP for planned configuration with normal work controls • ICCDP < 1E-6 • ICLERP < 1E-7 If exceeded compare to acceptance guidelines with risk management actions implemented to reduce sources of risk • ICCDP < 1E-5 • ICLERP < 1E-6	• Quantitative guidelines for normal work controls challenged, but acceptable with risk management actions implemented.

5.2 PRA QUALITY

The PRA quality has been assessed and determined to be adequate for this risk application, as follows:

- Scope - The BW PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA has the necessary scope to appropriately assess the pertinent risk contributors.
- Fidelity – The BW PRA model (BB011b4) is the most recent evaluation of the risk profile at BW for FPIE challenges. The PRA reflects the as-built, as-operated plant.

- Standards – The PRA has been reviewed against the ASME/ANS PRA Standard and the PRA elements are shown to have the necessary attributes to assess risk for this application.
- Peer Review - The PRA has received a peer review. Based on addressing the peer review results and subsequent gap analyses to the current standards, the PRA is found to have the necessary attributes to assess risk for this application.
- Appropriate Quality – The PRA quality is found to be appropriate to assess risk for this application.

5.3 QUANTITATIVE RESULTS VS. ACCEPTANCE GUIDELINES

As shown in Table 3.4-1 this analysis demonstrates with reasonable assurance that the proposed TS change is within the current risk acceptance guidelines in RG 1.177 for one-time changes. This combined with effective compensatory measures to maintain lower risk ensures that the TS change meets the intent of the ICCDP and ICLERP acceptance guidelines.

5.4 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the BW TS for the 2A Essential Service Water Pump to increase the CT from 72 hours to 200 hours when 2A SX is inoperable.

A PRA technical adequacy evaluation was also performed consistent with the requirements of ASME/ANS PRA Standard and RG 1.200, Revision 2. Additionally, a review of model uncertainty was performed with this application. None of these identified sources of uncertainty were significant enough to change the conclusions from the risk assessment results presented here.

However, the assessment of risk from internal events and internal fires did help to identify the following actions as important compensatory measures that will help to reduce the overall risk during the performance of the extended CT:

5.4.1 Compensatory Measures

- There will be no elective maintenance work on the remaining SX pumps (1A, 1B, 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.
- There will be no elective maintenance work on the emergency diesel generators (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 during the maintenance window.
- There will be no elective maintenance work on the Unit 2 auxiliary feed pumps (2A, 2B). This equipment will be protected for the one-time outage. This supports the maintenance assumptions in the risk analysis.
- 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.
- 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.

- There will be no elective maintenance on the startup feedwater pump, 2FW02P.
- There will be no elective maintenance activities on the Unit 2 Station Auxiliary Transformers.
- 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 loss of offsite power.
- Fire Risk Management Actions (RMAs) applicable for the 2A SX pump will be completed per OP-AA-201-012-1001 "OPERATIONS ON-LINE FIRE RISK MANAGEMENT" [Ref. 18] (these actions protect against fire impacting key redundant equipment)
- Operations will hold briefings on the following actions:
    - On a loss of all Reactor Coolant Pump (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 AF, 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 Residual Heat Removal (RHR) heat exchangers are used for ECCS 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 AF day tank from the 125K Fuel Oil Storage Tank in order to maintain operation of the diesel-driven AF pump.
  - On loss of Vital Instrument Bus (120 VAC) 211 or 214, Operations opens the AF 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 (MCR) to control SG levels.
- 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 walkdowns will be conducted to reduce and manage transient combustibles and to alert the staff about the increased sensitivity to fires in the following fire zones during the extended 2A SX outage window. Operating crew shift briefings will continue to be conducted every shift throughout the duration of the CT period. Additionally, planned hot work activities in the following fire zones will be prohibited during the time within the extended 2A SX 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 walkdowns 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)

Fire Zone[1]	Fire Zone Description
5.1-2	Division 22 ESF Switchgear Room
5.1-1	Division 12 ESF Switchgear Room
3.2-0	Auxiliary Building El. 439'-0"
11.4-0	Auxiliary Building General Area, El. 383'
11.6-2	Division 22 Containment Electrical Penetration Area, El. 426'
11.2C-2	Containment Spray Pump 2B Room
11.1B-0	Unit 2 Auxiliary Building Basement El. 330'
18.10D-2	Unit Auxiliary Transformer 241-2
18.10E-2	System Auxiliary Transformers 242-1/242-2

(2) For larger fire zones walkdowns may be focused on specific fire sensitive areas within the larger firezones. Walkdowns are judged as not being required for areas with continuous operator occupation (e.g. MCR). Fire Risk Management Actions (RMAs) where they occur may address the need for walkdowns in some of these areas. ALARA principles apply when reviewing radiological areas such as RHR.

6.0 REFERENCES

[1] Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," Revision 2, March 2009.

[2] 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, May 2011.

[3] Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1, May 2011.

[4] "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, Addenda RA-Sa-2003, and Addenda RA-Sb-2005, December 2005.

[5] "Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, February 2009.

[6] Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.

[7] BB-ASM-002, Application Specific Model Notebook RCP Shutdown Seals, Rev. 0, September 2015.

[8] BB PRA-014, "Byron/Braidwood PRA Quantification Notebook," Rev. BB011b, September 2012.

[9] LTR-RAM-II-13-067-NP, Rev. 0, "Byron / Braidwood Nuclear Plants RG 1.200 Internal Events and Internal Flooding PRA Peer Review Report", November 12, 2013.

[10] Exelon Risk Management Team, BW Fire PRA Fire Risk Quantification Notebook, BW-PRA-021.11, Rev. 0, May 2015.

[11] Exelon Risk Management Team, Development of Risk Management Actions for the Inclusion of Fire Insights into Braidwood Configuration Risk Management Program, BW-CRM-115, Rev. 1, February, 2014.

[12] BB-PRA-021.021.01, Revision 0, "Level 1 Seismic Quantification Notebook", December 2013

[13] "Individual Plant Examination of External Events For Severe Accident Vulnerabilities Submittal Report", June 1997, Braidwood Station

[14] NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 1991

[15] ER-AA-600, Risk Management, Revision 7.

[16] ER-AA-600-1015, FPIE PRA Model Update, Revision 18.

[17] USNRC, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," NUREG-1855, March 2009.

[18] Exelon, OP-AA-201-012-1001, "OPERATIONS ON-LINE FIRE RISK MANAGEMENT", Revision 1.

[19] Exelon, ER-AA-600-1046, "Risk Metrics – NOED and LAR", Revision 6.

[20] PWROG-14001-P, Revision 1, PRA Model for the Generation III Westinghouse Shutdown Seal, Risk Management Committee, PA-RMSC-0499R2, July 2014.

[21] PWROG-14006-P, Revision 0-B, Implementation Guide for the Generation III Westinghouse Shutdown Seal, Risk Management Committee, PA-RMSC-0499R1, December 2014.

[22] Westinghouse Technical Bulletin TB-15-1, Reactor Coolant System Temperature and Pressure Limits for the No. 2 Reactor Coolant Pump Seal, March 3, 2015.

[23] WCAP-16141, "RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs," August 2003.

[24] USNRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No 165 to Facility Operating License No. NPF-72 and Amendment No. 165 to Facility Operating License No. NPF-77," February 24, 2011