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RS-17-020 10 CFR 50.90

March 2, 2017

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket Nos. 50-461

Subject: Response to Request for Additional Information Concerning License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program," for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

References:

1. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program," for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated January 25, 2016 (ML16025A182)

2. Email from Jennie Rankin (U. S. NRC) to John L. Schrage (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 - Request for Additional Information regarding the License Amendment Request to Revise TS 5.5.13 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies (MF7290)," dated January 17, 2017 (ML17018A426)

3. Teleconference Between U. S. NRC (J. Rankin, et al) and Exelon Generation Company, LLC (J. Schrage, et al) on January 17, 2017

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License (FOL) No. NPF-62 for Clinton Power Station (CPS), Unit 1. The proposed amendment would revise FOL Appendix A, Technical Specification (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program" to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

In Reference 2, the NRC transmitted 13 requests for additional information (RAIs) related to the proposed license amendment. The NRC provided additional clarification of the RAIs during the Reference 3 teleconference.

EGC is providing the requested information, as described in Reference 2, in the Attachment and Enclosure to this letter.

There are no regulatory commitments contained within this letter.

If you have any questions concerning this letter, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of March 2017.

Respectfully,

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

Attachment: Response to Request for Additional Information, License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

Enclosure: RM Documentation No. CL-LAR-08, Revision 0, Response to RAIs Associated with ILRT LAR for Clinton Power Station

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

ATTACHMENT

Response to Request for Additional Information (RAI)
License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

Request for Additional Information (RAI) No. 1

Section 1 of Attachment 1 to the January 25, 2016, submittal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16025A182) requests revision to Clinton Power Station, Unit 1 (CPS) Technical Specifications (TS) 5.5.13 to:

> *[a]dopt a more conservative allowable test interval extension of nine months, for Type A, Type B and Type C leakage rate tests in accordance with NEI [Nuclear Energy Institute] 94-01, Revision 3-A.*

In a letter dated June 25, 2008 (ADAMS Accession No. ML081140105), the Nuclear Regulatory Commission (NRC) staff approved NEI 94-01, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2 (ADAMS Accession No. ML100620847). The NRC staff also identified in Section 3.1.1.2 that to extend Type A (i.e. integrated leak rate testing (ILRT)) beyond 15 years, the licensee must provide information demonstrating that there is an unforeseen emergent condition. Additionally, this 9-month extension is not to be used for routine scheduling or planning purposes. Given the requirement for demonstration of a non-routine emergent condition and that CPS operates on a 12-month refueling outage cycle, address the reasons and circumstances wherein the licensee expects to need such an extension.

Response to RAI No. 1

Exelon Generation Company, LLC (EGC) does not expect to need such an extension for CPS. The intent of the statement in the January 25, 2016 submittal (i.e., Section 1 of Attachment 1) was made to acknowledge the change in allowable extension from 15 months, as delineated in NEI 94-01, Revision 0 (i.e., the version of the standard which currently governs the CPS containment leak rate testing program), to that of 9 months, as delineated in NEI 94-01 Revision 3-A.

Except for compelling reasons, which could include unforeseen emergent conditions, EGC will conduct the Type A tests within the approved 15-year interval, without seeking extensions. If an unforeseen emergent condition should arise, extension of the Type A test interval will be addressed in accordance with the June 25, 2008 NRC letter and Safety Evaluation that approved NEI 94-01 Revision 2, and Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J of 10 CFR Part 50."

RAI No. 2

The following apply to the internal events Facts & Observations (F&Os) reported in Attachment 1 to the March 31, 2016 supplement (ADAMS Accession No. ML16091A077):

a. *F&Os 1-3, 1-26 and 6-8 found that room heat-up calculations were not performed to support the exclusion of room ventilation from the PRA model. The resolutions of these F&Os discuss room heat-up calculations performed for the switchgear rooms and the residual heat removal (RHR) 'B' room. Address whether these are the only rooms in the*

plant that the cited F&Os apply to and whether there are no other risk-significant rooms where loss of ventilation could cause equipment failure.

b. *The resolutions of F&Os 1-26 and 6-8 address the rooms cited in the F&Os (e.g. the main control room and several emergency core cooling system (ECCS) rooms) using evaluations for other rooms and qualitative arguments. Address how the calculations performed for a particular room are applicable to other rooms. The justification can include information such as volume and heat load comparisons.*

c. *For F&O 1-4, related to supporting requirement (SR) IE-A6, there was no evidence that a systematic evaluation of initiating events due to multiple equipment failures and routine system alignments has been performed. Address whether this systematic evaluation was performed and discuss whether any new initiating events have been identified.*

d. *In F&O 1-14, related to SR IFSN-A6, the peer review team found no evidence that a systematic assessment of the effects of jet impingement, pipe whip, humidity, temperature, etc. on systems, structures and component s (SSCs) was performed. Describe the systematic assessment of these internal flooding effects and summarize its conclusions.*

e. *In F&O 1-17, related to SR IE-C3, it is stated that the basis for recovery actions for certain initiating event fault trees was not documented. Similarly, F&O 1-21, related to SR AS B3, states that no evaluation for the ECCS pump operation at post-containment venting conditions was provided. The licensee's dispositions of these F&Os indicate that the basis is documented in various plant-specific reports. Provide a discussion of the basis for crediting the recovery actions mentioned in F&O 1-17 and for crediting ECCS operation after containment venting mentioned in F&O 1-21.*

f. *For F&O 1-24, which is related to SR SY-A18, identifies the lack of a concerted effort to identify accident conditions that could cause system failures. F&O 1-22, related to SR AS-B3, also raises a similar concern. Briefly describe the process followed to identify, review, and model accident conditions that could cause system failures.*

g. *For F&O 1-27, which is related to SR SY-A24 and DA-C15, it was found that plant-specific data was not analyzed to support crediting RHR repair in the probabilistic risk assessment (PRA) model. The resolution to this F&O states that the repair data are based on industry experience as allowed by SR DA-C15 and are judged to be reasonable. The NRC staff's endorsement of the 2009 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009 in Regulatory Guide (RG) 1.200, Revision 2, includes the exception for SR DA-C15, which states that industry experience should be used only if plant-specific experience is insufficient to estimate the failure to repair. Explain why the plant specific experience is insufficient to estimate the failure to repair.*

h. *For F&O 1-43, which is related to SR QU-D1, it is stated that a review of the "significant" cutsets and accident sequences as defined in the standard was not performed. The resolution states that the top cutsets and accident sequences now represent a larger percentage of the total core damage frequency (CDF) compared to the time of the peer-review, but does not address the concern in the F&O. Similarly, F&O 1 46, related to SR QU-D5, states that a review of the non-significant cutsets and accident sequences was*

> *not performed. The resolution for F&O 1-46 contains a reference to a licensee guidance document, but does not address the concern of the F&O. Address whether a review of significant cutsets and accident sequences, as defined in the 2009 ASME/ANS PRA Standard RA-Sa-2009, as well as non-significant cutsets and accident sequences was performed.*

Response to RAI No. 2

Response to this RAI is provided in the Enclosure.

RAI No. 3

In the March 31, 2006 supplement, resolution of F&O 5-7 states:

> *[t]he [Electric Power Research Institute] EPRI [Human Reliability Analysis] HRA Calculator is now used to quantify the probabilities of the [Human Error Probabilities] HEPs in the Clinton model...The transition to using the EPRI HRA Calculator did not represent a methodology change...*

In addition, resolutions of all F&Os related to the human reliability (HR) supporting requirements (F&Os 1-31, 1-32, 1-33, 1-34, 2 16, 3-13, and 5-10) seem to indicate that many HRA related changes to the PRA model were performed in the 2011 PRA update and/or as a result of using the EPRI HRA Calculator.
The 2009 ASME/ANS PRA Standard RA-Sa-2009 defines a PRA upgrade as,

> *...incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences.*

Non-mandatory Appendix 1-A of the PRA Standard cites "a different HRA approach to human error analysis..." as a potential PRA upgrade. Based on Section 1-5, "PRA Configuration Control," of the standard and RG 1.200, Revision 2, Regulatory Position 1.4, "PRA Development, Maintenance, and Upgrade," a PRA upgrade must be peer reviewed.

a. *Justify why this model change is not considered a PRA upgrade requiring a focused-scope peer review. If this change qualifies as an upgrade, provide the results from of the focused-scope peer review addressing the associated F&Os and their resolutions.*
b. *Provide an overview of all other changes in the internal events PRA model that occurred after the 2009 peer review, and clarify whether any of these changes qualify as a PRA upgrade that would require a focused-scope peer review.*

<u>Response to RAI No. 3</u>

Response to this RAI is provided in the Enclosure.

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Response to Request for Additional Information (RAI)
License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

RAI No. 4

One of the topical report use conditions identified in the safety evaluation (SE) approving the methodology in EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105) is that the average leak rate for the pre-existing containment large leak class (i.e., Class 3b) be assigned a value of 100 times the maximum allowable containment leakage rate (La) instead of the previously used value of 35 La. Due to the methodology used, the pre-existing drywell large leak also uses the new postulated leakage rate (i.e., 100 times the maximum allowable drywell leakage rate [DWLb]). These leakage rates (i.e., 100 La and 100 DWLb) represent a substantial increase from the values used for the previous one-time frequency extension request (ADAMS Accession No. ML030370524).

In Section 4.1.3 of Attachment 4 to the submittal, the licensee refers to the drywell bypass leakage rate test (DWBT) risk assessment methodology used for its previous one-time ILRT/DWBT extension request as the basis for the approach used. In the same section, the licensee discusses the impact of drywell leakage on containment over-pressurization and CDF based on the margin found in the results of the deterministic calculations documented in the previous one-time ILRT/DWBT frequency extension request. However, discussion of the impact of the increased DBWT leakage of 100 DWLb on the amount of Cesium Iodide (CsI) released is not provided. Therefore, address why the increased DWBT leakage of 100 DWLb does not change the assignment to the EPRI accident Class 3a and Class 3b shown in Table 4.6-1 of Attachment 4 to the submittal.

Response to RAI No. 4

Response to this RAI is provided in the Enclosure.

RAI No. 5

Similar to the previous one-time ILRT/DWBT frequency extension request, summarize the results of the sensitivity analysis including the probability of drywell failures assigned to small (Class 3a) or large (Class 3b) drywell bypass (DWB) leakage, using the current "as-found" DWBT leakage data for all Mark III containments. Provide the changes (i.e., delta) in large early release frequency (LERF), population dose, conditional containment failure probability, and impact on baseline LERF. The NRC staff notes that the sensitivity documented in Section 6.3 of Attachment 4 to the submittal which increases the probability values of the small (Class 3a) and large (Class 3b) DWB leakage by a factor of 10, does not appear to capture the corresponding probabilities determined based on the historical DWB leakage data using the Chi-square upper bound value.

Response to RAI No. 5

Response to this RAI is provided in the Enclosure.

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Response to Request for Additional Information (RAI)
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RAI No. 6

Section 5.7 of Attachment 4 to the submittal, states that other external hazards, including high winds and tornadoes, external floods, transportation accidents, and nearby facility accidents were not considered because of their negligible contribution to overall plant risk. This conclusion was reached based on the CPS Individual Plant Examination for External Events (IPEEE) analysis. Since the IPEEE studies have not been recently updated, discuss the applicability of the IPEEE conclusions with regards to each of the above-mentioned hazards to the current plant configuration and operating experience, and taking into account updated risk studies and insights.

Response to RAI No. 6

Response to this RAI is provided in the Enclosure.

RAI No. 7

The licensee has discussed its Fire PRA in Section A.3.1 in Appendix A of Attachment 4 to the submittal. Address the following regarding the fire PRA:

a. *Address, preferably quantitatively such as through sensitivity analyses, whether the estimated fire CDF (6.0E-6/yr) and LERF (9.21E-7/yr) are bounding with respect to the current state of the-art for fire PRA considering all approved guidance since NUREG/CR-6850 was first issued.*
b. *Address whether any "unapproved/unreviewed analysis methods" were employed in the current application of the Fire PRA.*

Response to RAI No. 7

Response to this RAI is provided in the Enclosure.

RAI No. 8

Section 5.3 in Attachment 4 to the submittal, states that Class 7 sequences are impacted by the ILRT/DWBT interval extension. The statements appears to be inconsistent with arguments in Section 4.1.3, which explains that Class 7 sequences are not impacted by the requested extension. As Tables 5.2-2, 5.3-1 and 5.3-2 do not show any impact on Class 7, clarify the treatment of Class 7 sequences.

Response to RAI No. 8

Response to this RAI is provided in the Enclosure.

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Response to Request for Additional Information (RAI)
License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

RAI No. 9

Table 3.4.4-1 "CPS Type A ILRT History" of Attachment 1 to the submittal, provides the details of the historical ILRT "As-found Leakage Rate" and "As-Left Leakage Rate" values. Section 9.2.3 "Extended Test Intervals" of Nuclear Energy Institute (NEI) 94-01 Revision 2-A states that:

> *[i]n the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure (Pa).*

Provide the actual ILRT test pressures employed during the two most recent Type A tests.

Response to RAI No. 9

The actual ILRT test pressures employed during the two most recent Type A tests are as follows:

November 1993 - Final Type A Pressure,	9.474 psig
February 2008 - Final Type A Pressure	9.5758 psig

RAI No. 10

In Table 3.8.1-1 "NEI 94-01 Revision 2-A Limitations and Conditions" of Attachment 1 to the submittal, the licensee indicates that:

> *CPS will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.*

Table 3.4.4-1 of Attachment 1 to the submittal provide the details of the historical ILRT Containment "As-found Leakage Rate" and "As-Left Leakage Rate" values. With respect to Table 3.4.4-1:

a. *Provide the definition of "performance leakage rate" used during the February 2008 [ILRT].*

b. *From the Table 3.4.4-1 of ILRT results, it can be seen that the "Leakage Rate" increased by 22.9% [0.2708 "As found"/ 0.2204 "As Left"] between the time the ILRT of November 1993 was performed and when the ILRT of February 2008 was performed. Discuss the cause of the increase in containment leakage rates.*

Response to RAI 10

a. *Provide the definition of "performance leakage rate" used during the February 2008 [ILRT].*

The 2008 ILRT was performed in accordance with CPS Procedure 9861.01, "Integrated Leakage Rate Test." This procedure was written and verified to be in compliance with the applicable sections of:

- 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B
- NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program"

- NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J"
- ANS/ANSI 56.8-1994, "Containment System Leakage Testing Requirements"

At the time of the 2008 ILRT, Clinton was licensed as follows:

> A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions: (1) Bechtel Topical Report BN-TOP-1 is also an acceptable option for performance of Type A tests, and (2) NEI 94-01 - 1995, Section 9.2.3: The first Type A test performed after November 23, 1993 shall be performed no later than November 23, 2008.

NEI 94-01 Revision 0 defined, for purposes of determining an extended test interval, the performance leakage rate as follows:

> For purposes of determining an extended test interval, the performance leakage rate is determined by summing the UCL (determined by containment leakage rate testing methodology described in ANSI/ANS 56.8-1994) with As-left MNPLR leakage rates for penetrations in service, isolated or not lined up in their accident position (i.e., drained and vented to containment atmosphere) prior to a Type A test. In addition, any leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. If the leakage can be determined by a local leakage rate test, the As-left MNPLR for that leakage path must also be added to the Type A UCL. If the leakage cannot be determined by local leakage rate testing, the performance criteria for the Type A test are not met.

The acceptance criteria as stated in the surveillance procedure for the 2008 ILRT was as follows:

> The upper bound of the leakage rate (UCL) calculated at 95% confidence level, based on mass point calculations, is less than 0.75 La (0.75 x 0.65 = 0.4875 wt.%/day) as recorded in this test. Any required local leak rate additions for leakage paths that have to be blocked for successful testing (Post-Repair), or for systems that should be vented to the containment but are not will be added to the UCL and documented in CPS 9861.01D002.

b. From the Table 3.4.4-1 of ILRT results, it can be seen that the "Leakage Rate" increased by 22.9% [0.2708 "As found"/ 0.2204 "As Left"] between the time the ILRT of November 1993 was performed and when the ILRT of February 2008 was performed. Discuss the cause of the increase in containment leakage rates.

The February 2008 ILRT included an As-found penalty of 0.064 wt%/day for penetrations that were isolated during the test. This factor is the most significant contributor to the 22.9%

There were two penetrations that made up the majority of the 0.064 wt%/day, with a total contribution of 86%. Their As-found and As-left contributions are listed in the table below:

Penetration Leakage Added to ILRT			
Penetration	Description	As-found Minimum Pathway [1] (sccm)	As-left Minimum Pathway (sccm)
1MC-016	RHR B Injection	4450.0	3528
1MC-045	MSL Drain Line	15,600	200
[1] penetration repaired; this number represents as-found Minimum Pathway leakage before repair.			

RAI 11

In Table 3.8.1-1 of Attachment 1 to the submittal, the licensee indicates that "...there no major modifications planned." Section 9.2.4 of NEI TR 94-01, Rev. 2, indicates that Type A testing is required after major modifications to the containment or upon approval by the NRC the licensee may perform a short duration structural test of the containment. For minor modifications or modifications to the pressure boundary an LLRT was indicated. As the Unit 1 containment has been in service for approximately 28 years, provide a summary of all significant modifications to the Unit 1 containment and the subsequent post modification testing. The summary should discuss the extent to which actions were completed consistent with the NRC staff approval of NEI TR 94-01, Rev. 2 dated June 25, 2008 (ADAMS Accession No. ML081140105).

Response to RAI 11

Section 9.2.4 of NEI TR 94-01, Revision 2, states that:

> Repairs and modifications that affect the containment leakage integrity require LLRT or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation.

Article IWE-5000 of the ASME Code, Section XI, Subsection IWE (up to the 2001 Edition and the 2003 Addenda), would require a Type A test after major repair or modifications to the containment. In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure.

EGC has not made any modifications to the CPS primary containment since the last ILRT was performed in 2008 and only one in the history of the unit. EGC reviewed all CPS ILRT Summary Reports from C1R01 thru C1R16 (i.e., the 2016 refueling outage). This search identified one engineering change (ECN) that was performed during C1R05 (i.e., the 1995 refueling outage). This ECN implemented a modification to the spare containment penetration 1MC-074 to provide a permanent access connection for use during chemical decontamination activities. The work consisted of removal of the blank head fittings and installation of a new head fitting, pipe penetration, support, and provisions for LLRT testing. An LLRT was performed on this penetration as a post modification test.

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Response to Request for Additional Information (RAI)
License Amendment Request to Revise Technical Specification Section 5.5.13, "Primary Containment Leakage Rate Testing Program" for Permanent Extension of Type A and Type C Leak Rate Test Frequencies

RAI 12

Table 3.6.5-1, "CPS Type B and C LLRT Program Implementation Review" of Attachment 1 to the submittal, identified components that were on extended intervals and have not demonstrated acceptable performance for Type C testing during the previous two outages (C1R14-2013, and C1R15-2015). Based on the footnotes for the table, address whether the "As-found SCCM" for the three components in the table are "As-found maximum SCCM" for those penetrations and the "As-left SCCM" for the same components are "As-left maximum SCCM" for the corresponding penetrations. If any of the "As-found SCCM" in Table 3.6.5-1 are "As-found minimum SCCM", address whether they are included in Table 3.6.4-1 "As-Found min path (SCCM)."

Response to RAI No. 12

The leakage rate values listed in Table 3.6.5-1 are actual component LLRT test results.

As-found and As-left maximum and minimum pathway values (i.e., MXPLR and MNPLR, respectively) for each of the components and penetrations in Table 3.6.5-1 are provided in a revised version of Table 3.6.5-1, which follows:

Revised Table 3.6.5-1, CPS Type B and C LLRT Program Implementation Review

C1R14 – 2013						
Component (Penetration)	**As-found SCCM**	**Evaluation Limit SCCM**	**As-left SCCM**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
1IA175 Instrument Air Isolation Check Valve to 1IA006 (1MC057)	MXPLR 56000 (1) MNPLR 704.25	500	MXPLR 1680 MNPLR 704.25	Seat leakage	Replaced Valve	30 months
C1R15 – 2015						
Component (Penetration)	**As-found SCCM**	**Evaluation Limit SCCM**	**As-left SCCM**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
1RF021 Containment Building floor Drain Inboard Isolation Control Valve (1MC070)	MXPLR 22000 MNPLR 130	10000	MXPLR 695 MNPLR 130	Intermediate position indication (2)	Refurbished actuator (2)	30 months
1VR006B Continuous Containment Purge Inboard Isolation Valve (1MC113)	MXPLR 24980 MNPLR 3250	500	MXPLR 6500 MNPLR 3250	Seat leakage (3)	Replaced valve. Accepted As-left above admin limit by eval.	30 months

(1) 1IA175 failed the As-found LLRT with 56000 sccm. However, the As-found, Minimum Pathway, leakage for penetration 1MC057 was 704.25 sccm. As-left value of 1680 sccm was determined to be acceptable.

(2) 1RF021 indicated intermediate when attempting to close. When the test volume was pressurized to test pressure, air flow was felt on the outlet of the test vent valve 1RF030B. This intermediate indication was the most probable cause of the excessive leakage. Resolved intermediate position indication in 1RF021 and re-tested.

(3) 1VR006B failed the as-found LLRT with 24980 sccm. This is added to the Max-Path leakage. This is also a secondary containment bypass penetration so this leakage applies to both the $0.6L_a$ total and the $0.08L_a$. The previous test results for this penetration was 6,904 sccm. Following maintenance, the As-left Maximum Pathway leakage for penetration 1MC113 was tested as 6500 sccm. This value is 94.2% of the 1MC113 As-left Max Pathway leakage measured in C1R12 which had a value of 6900 sccm (6500 / 6900 = 94.2%). The leakage identified for 1MC113, was determined to be an improvement of past leakage for the penetration and secondary containment bypass as a whole. Because of this improvement, EGC accepted the As-found test results of 1MC113 and the impact on $0.08L_a$ (secondary containment bypass), and $0.6L_a$ (containment leakage). No further action was required.

RAI No. 13

Like other Mark III containments, Unit 1 has an internal drywell interfacing with the surrounding primary containment through a suppression pool. When a loss-of-coolant accident occurs, the drywell atmosphere pushes down on the suppression pool displacing it past the weir wall into the drywell to the volume under the hydraulic control unit (HCU) floor. The surge of water and drywell atmosphere gas rises to the HCU floor and given the area restriction of the floor, a short term differential pressure is developed from below (wetwell space) to above the floor. When Pa is calculated describe which region (i.e. above or below HCU floor) is credited. If the Pa is not equal to or conservative relative to the peak pressure below the HCU floor, provide the peak calculated DBA-LOCA internal pressure in the wetwell space, justify the calculated Pa value and the acceptability of performing ILRT/LLRT at a lower pressure than would occur in the wetwell space.

Response to RAI No. 13

As stated in CPS Technical Specification Bases B3.6.1.1, "Primary Containment," the Appendix J test pressure is 9.0 psig. This value was chosen to bound the calculated peak pressure in containment during a DBA-LOCA of 8.74 psig, as originally licensed. The original licensing basis value of 8.74 psig is the peak pressure in the containment for the long-term transient response. For the long-term response, pressure will be equalized between the containment and drywell volumes. The original licensing basis peak pressure in the wetwell during the initial short-term transient response (i.e., the first 30 seconds of the transient) is 7.65 psig.

In 2002, EGC revised the DBA-LOCA analysis as part of Extended Power Uprate (EPU). This revision determined that the peak containment pressure for the long-term transient response is

6.97 psig, while the peak pressure in the wetwell during the short-term transient is 9.22 psig. The revised values are due to both changes in the initial conditions and modeling changes.

The EPU analysis states that since the peak pressure of 9.22 psig is short-lived and occurs only in the wetwell region, the peak short-term wetwell pressure is not used to define the containment leakage test pressure. The peak containment pressure is derived from the long-term transient response. With respect to the long-term transient response, although the NSSS vendor recommended the use of 7 psig as the containment test pressure for Appendix J Program (i.e., to bound the calculated value of 6.97 psig), EGC decided to continue to use 9 psig as a conservative value. This issue was discussed in EGC's LAR for EPU (i.e., letter from J. M. Heffley (AmerGen) to U. S. NRC dated June 18, 2001; ADAMS Accession No. ML011720516). The LAR was approved by the NRC in a Safety Evaluation dated April 5, 2002 (ADAMS Accession No. ML021650543).

Enclosure

RM Documentation No. CL-LAR-08 Revision 0

Response to RAIs Associated with ILRT LAR for Clinton Power Station

Exelon.

Risk Management Team

RM DOCUMENTATION NO. CL-LAR-08	REV: 0	PAGE NO. 1

STATION: Clinton Power Station
UNIT(s) AFFECTED: 1

TITLE: RAI Response for ILRT LAR for Clinton Power Station

SUMMARY:

The purpose of this document is to provide Exelon Generation's responses to Requests for Additional Information (RAIs) associated with Clinton ILRT License Amendment Request.

This is a Category I RM Document IAW ER-AA-600-1012, which requires independent review and approval.

[ ] Review required after periodic Update

[X] Internal RM Documentation [ ] External RM Documentation

Electronic Calculation Data Files: N/A

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

This RM documentation supersedes: N/A

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RESPONSE TO RAIs ASSOCIATED WITH ILRT LAR FOR CLINTON POWER STATION

RAI 2a

Question

F&Os 1-3, 1-26 and 6-8 found that room heat-up calculations were not performed to support the exclusion of room ventilation from the PRA model. The resolutions of these F&Os discuss room heat-up calculations performed for the switchgear rooms and the residual heat removal (RHR) 'B' room. Address whether these are the only rooms in the plant that the cited F&Os apply to and whether there are no other risk-significant rooms where loss of ventilation could cause equipment failure.

Response to RAI 2a

A nuclear power plant has many rooms, with a noteworthy proportion containing equipment explicitly or implicitly modeled in the PRA. However, it is not necessary to model all room cooling methods, or perform heatup evaluations for all these rooms. What is important is to capture the key room cooling dependencies in the model (that is where equipment failure is realistic if a designed cooling system fails). That is the essence of the associated supporting requirements from the ASME PRA Standard (e.g. SRs SY-A22, SY-B6, SY-B7). If a room or piece of equipment has a special cooling system by design this may point to its consideration in the model. Clinton has modeled these cooling dependencies, where it is judged to be most critical, that is where there is high heat load or where the natural heat sinks (e.g. concrete in the walls) for heat dispersion are small. For example the room coolers for ECCS Pump rooms and Shutdown Service Water Pump rooms (Divs 1 and 2) have been modeled. Where room cooling was modeled, there generally are design basis calculations in support of these design basis cooling sources.

Where there are not significant heat loads (e.g. cable spreading room) or where the volumes are large relative to the heat load in the room (.e.g. the Screenhouse area housing the plant service water pumps) no room cooling was modeled (and in fact there may be no special room cooling feature in the plant design, indicating that success can be achieved without room cooling.)

Some areas of the plant have room cooling features, but the room cooling feature was judged as not being required to maintain equipment functionality for the PRA mission time (e.g. the switchgear rooms, which are cooled by Essential Switchgear Heat Removal). The judgment regarding switchgear heat removal not being required was an original assumption of the Clinton PRA model and is judged a reasonable assumption given that these type of switchgear are used in many industrial settings, sometimes in relatively harsh environments. The essence of the peer review findings was that a

better analytical basis should have been provided especially in areas where it would not be immediately obvious where success could be achieved without cooling.

Table 2-1 summarizes the cooling dependencies modeled for safety related equipment modeled in the PRA. The equipment in Table 2-1 compose the noteworthy risk significant systems, when considering room cooling dependencies. Non-safety related equipment such as feedwater and the main condenser, which are part of the power conversion systems, are not shown here as they generally have lower risk significance than the safety related equipment. Much of this power conversion equipment is directly cooled by water cooling systems, such as Turbine Building Closed Cooling Water, Component Cooling Water or Plant Service Water, which have modeled dependencies. Furthermore to the extent that HVAC systems could contribute to a loss of Feedwater or Main Condenser, they would be addressed by the Loss of Feedwater or Loss of Main condenser initiators already existing in the model.

As indicated in Table 2-1, room cooling is not modeled as required for the following risk significant rooms:

- Switchgear Rooms
- Cable Spreading Rooms and cable routing areas
- Battery Rooms
- Main Control Room
- Division 3 SX Pump Room
- RCIC Room for 4 hour SBO mission time

Justification for not requiring room cooling is included in Table 2-1.

TABLE 2-1

ROOM COOLING DEPENDENCIES FOR SAFETY RELATED EQUIPMENT MODELED IN THE CLINTON PRA

TYPE OF SYSTEM	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
Front Line Emergency Core Cooling System (Motor Driven)	LPCS and RHR A ECCS room coolers are modeled which are supplied by Div 1 Shutdown Service Water (SX) Open Pump room doors are also regarded as providing acceptable cooling as a backup to the room coolers.[1]	RHR B and RHR C ECCS room coolers are modeled which are supplied by Div 2 Shutdown Service Water (SX) Open Pump room doors are also regarded as providing acceptable cooling as a backup to the room coolers.[1]	HPCS ECCS room coolers are modeled which are supplied by Div 3 Shutdown Service Water (SX) Open Pump room doors are also regarded as providing acceptable cooling as a backup to the room coolers.[1]	NA
Reactor Core Isolation Cooling (Steam Driven)	RCIC 24 hour mission time ECCS room coolers are modeled which are supplied by Div 1 SX. Opening pump room doors is also regarded as providing acceptable room cooling as a backup to the room cooler.[(1)] RCIC 4 hour SBO mission time, no room cooling is required based upon SBO analysis.	NA	NA	NA
Shutdown Service Water (SX) (which provides Cooling Water to divisional cooling loads)	Div 1 SX Pump Room Cooling is modeled which is provided with water from the Div 1 SX pump.	Div 2 SX Pump Room Cooling is modeled which is provided with water from the Div 2 SX pump	Div 3 SX Pump Room cooling is not modeled for this room as the Div 3 SX Pump is a much smaller pump than Div 1 or 2, leading to a lower heat load.	NA

TABLE 2-1

ROOM COOLING DEPENDENCIES FOR SAFETY RELATED EQUIPMENT MODELED IN THE CLINTON PRA

TYPE OF SYSTEM	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
Emergency Diesel Generators	Div 1 diesel is primarily cooled by jacket water cooling provided from Div 1 SX, this dependency is modeled. Also diesel generator HVAC is modeled for bringing in outside air for room cooling. Both are required in the model for success.	Div 2 diesel is primarily cooled by jacket water cooling provided from Div 2 SX, this dependency is modeled. Also diesel generator HVAC is modeled for bringing in outside air for room cooling. Both are required in the model for success.	Div 3 diesel is primarily cooled by jacket water cooling provided from Div 3 SX, this dependency is modeled. Also diesel generator HVAC is modeled for bringing in outside air for room cooling. Both are required in the model for success.	NA
Switchgear Rooms	Switchgear Heat Removal not modeled because of relatively low heat load to room volume.[2]	Switchgear Heat Removal not modeled because of relatively low heat load to room volume.[2]	Switchgear Heat Removal not modeled because of relatively low heat load to room volume.[2]	NA
Nuclear System Protection System Inverter rooms	Switchgear Heat Removal (several methods available) is modeled. Safety related has dependency on on Div 1 SX. Non-safety depends on balance of plant cooling sources.	Switchgear Heat Removal (several methods available) is modeled. Safety related has dependency on on Div 2 SX. Non-safety depends on balance of plant cooling sources.	Switchgear Heat Removal (several methods available) is modeled. Safety related has dependency on on Div 3 SX. Non-safety depends on balance of plant cooling sources.	Switchgear Heat Removal (several methods available) is modeled. Div 4 inverter can be cooled by Div 2 SX and other methods.
Cable Spreading Rooms and cable routing areas.	Heat removal not modeled. Heat load tends to be low relative to room size, and cables capable of functioning at higher temperatures.[3]	Heat removal not modeled. Heat load tends to be low relative to room size, and cables capable of functioning at higher temperatures.[3]	Heat removal not modeled. Heat load tends to be low relative to room size, and cables capable of functioning at higher temperatures.	Heat removal not modeled. Heat load tends to be low relative to room size, and cables capable of functioning at higher temperatures.

TABLE 2-1

ROOM COOLING DEPENDENCIES FOR SAFETY RELATED EQUIPMENT MODELED IN THE CLINTON PRA

TYPE OF SYSTEM	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
Battery Rooms	Heat removal not modeled. Batteries are designed to handle temperature increases due to discharge and there is little else in the room.	Heat removal not modeled. Batteries are designed to handle temperature increases due to discharge and there is little else in the room.	Heat removal not modeled. Batteries are designed to handle temperature increases due to discharge and there is little else in the room.	Heat removal not modeled. Batteries are designed to handle temperature increases due to discharge and there is little else in the room.
Main Control Room	MCR HVAC Div 1 is not modeled because of the relatively large size of the MCR to its heat load.[4]	MCR HVAC Div 2 is not modeled because of the relatively large size of the MCR to its heat load.[4]	NA	NA

Notes to Table 2-1:

(1) The efficacy of door opening as a backup method for cooling of ECCS pump rooms is based upon a Computational Fluid Dynamics (CFD) analysis of RHR B Pump and Heat Exchanger Room using FLUENT. This demonstrates that the natural air flow through open doors driven by air density effects is sufficient to limit the room temperatures near or below equipment qualification temperatures. The other ECCS rooms are comparable, although not identical. However, the analysis for RHR B room shows that open doors do have a significant cooling impact capable of limiting peak room temperatures. The RHR A room is judged as having similar behavior to RHR B because the rooms are generally symmetrical in terms of dimensions and heat loads. To address the uncertainty of this similarity argument, to other ECCS rooms and the RCIC room, there is a phenomenological failure term in the model that represents the possibility that these similarity arguments may not apply as well to the other ECCS rooms (i.e. LPCS, RHR C, HPCS and RCIC). The phenomenological term has a value of 0.1 with a RAW of ~1.1.

(2) Room heatup analysis has been performed for the Division 1 and Division 2 Switchgear Rooms using GOTHIC to understand expected temperature increases under design basis conditions, but without the Switchgear Heat Removal system available for heat removal. Without any room cooling room temperatures are shown to be below equipment temperature limits for the first 10 hours with no action. Furthermore, temperatures remain below equipment qualification temperatures for over 24 hours (the PRA mission time) if the Div. 1 room doors are opened at 10 hours. Temperatures in the Div. 2 room switchgear room will remain below the equipment qualification temperatures for at >24 hours if doors remain closed. Door opening is proceduralized and is triggered by temperature alarm procedures. Placement of temporary fans extends this duration indefinitely. This analysis using design basis inputs is regarded as being conservative in several ways, with maximum assumed heat loads and maximum ambient temperatures. Furthermore equipment failure is not expected to occur right at the EQ temperature limit. Based upon this analysis, it is concluded that Switchgear Room Cooling likely would not contribute to equipment failure if it were to fail during an event, and does not introduce new dependencies for the associated divisional buses. Any temporary fans placed could be powered by the associated divisional bus.

(3) Room heatup analysis has been performed for the Division 1 and Division 2 Cable Spreading Rooms and adjoining cable routing areas using GOTHIC to understand expected temperature increases under design basis conditions, but without a HVAC system available for heat removal. Without any room cooling room temperatures remain below 122°F for the first 24 hours for all rooms.

(4) The Main Control Room air envelope is large occupying almost the entire floor area of the 800" elevation of the Control Building. The MCR envelope was designed to be big enough to accommodate a dual unit control room, while equipment was installed only for one. While it has numerous electrical panels the heat load per floor area is roughly comparable to the safety related switchgear rooms in the plant, which do not have heat removal modeled either (see footnote 2 above). Unlike the switchgear rooms the adjacent areas are not hot from piping containing steam or hot water; in fact most of the main control room perimeter is exposed to outside air, which would limit transmission heat loads into the MCR as it heats up if it were to lose cooling. The MCR has provisions for exhausting warm air from the MCR in the event of a total loss of MCR cooling (as would happen during a Station Blackout). This is achieved through the use of a temporary engine driven exhaust fan, which pulls air from the MCR panel area through a vent hose and exhausts it outside the building. Because the Main Control Room is continually occupied by operators, it is assumed that they would take actions as necessary to maintain temperatures habitable including use of this temporary equipment as needed.

RAI 2b

Question

The resolutions of F&Os 1-26 and 6-8 address the rooms cited in the F&Os (e.g. the main control room and several emergency core cooling system (ECCS) rooms) using evaluations for other rooms and qualitative arguments. Address how the calculations performed for a particular room are applicable to other rooms. The justification can include information such as volume and heat load comparisons.

Response to RAI 2b

The modeling approach used for room cooling dependencies for divisional equipment (generally some of the most PRA significant equipment in the plant) is discussed in the response to question 2a above. The approach for room cooling modeling should be understood in consideration of the objectives of the PRA model. The objective is to present a risk characterization that is as realistic as possible, without either undue conservative or optimistic bias. Exact knowledge of room cooling dependencies is generally imperfect because existing design calculations typically do not account for non-steady state heat transfer and mass transfer effects without coolers available, and there is a lack of knowledge regarding the temperature at which a piece of equipment is expected to fail. It would be reasonable to conclude that if a piece of equipment is kept below its Equipment Qualification Temperature, that its failure rate would not increase appreciably. However, it would not be typical for a piece of equipment to immediately fail once its Equipment Qualification Temperature has been exceeded. Equipment Qualification Temperature limits are typically confirmed by testing or analysis, and are not necessarily maximum temperatures at which equipment will function. Unless the melting point of some material is exceeded or the thermal stresses exceed possibly reduced thermal stress capacities, failure would not necessarily be prompt. Higher temperatures can cause insulation to break down and lubricants to change properties, which would in turn lead to higher failure probabilities, but failure is not certain, especially in the near term.

As a result a certain amount of judgment has been exercised in the modeling decisions for room cooling dependencies, especially in areas where it is known that the heatup rates would be low and that recovery would be likely. The room heatup analyses that are available show the efficacy of heat removal through open doors. The air flow through open doors is driven by air density differences, where the flow rate tends to increase as the differential temperature increases, helping to improve the heat removal rate.

ECCS Pump Rooms

The room cooling modeling scheme for ECCS rooms is as follows:

- All rooms have their room coolers provided by the plant design modeled as a means of success.
- ECCS room door opening, is allowed by high room temperature annunciator procedures for the rooms, in the event the ECCS is needed for core cooling. The door opening operator action is modelled as a backup to the room cooler. The efficacy of door opening was demonstrated through a Computational Fluid Dynamics (using FLUENT) analysis of the RHR B Pump and Heat Exchanger room. When doorways to the exterior of this room are opened, airflow through the doors was sufficient to keep air temperatures at or below Equipment Qualification temperatures. Due to the nearly symmetrical nature of the RHR A pump and Heat Exchanger room, this same modeling was applied to the room cooling model for these rooms. Note the ECCS rooms tend to have fairly high environmental qualification temperature envelopes to accommodate high energy line breaks in nearby compartments.
- HPCS, LPCS and RHR C have somewhat bigger design heat loads, especially for the core spray systems due to the somewhat larger pump motors. But these rooms are also larger and have more concrete area for an energy heat sink. Because the FLUENT analysis for RHR B demonstrated the efficacy of door opening for dispersing heat energy, door opening is believed to be a successful strategy for room cooling for these rooms as well. Therefore ECCS room door opening is applied as a backup cooling method for these rooms too. Note only one door opening operator action basic event is used in all cases, to account for the dependency between these door opening actions. Because the room geometries and heat loads are somewhat different than the RHR A and B rooms, these rooms also have a phenomenological term with a probability of 0.1 and RAW of ~1.1, applied (ORed with the door opening action) to address the possibility that they are different enough that the door opening is not sufficient. The phenomenological term is a common cause term conservatively impacting multiple rooms, assuming each room's HVAC is unavailable.
- RCIC room cooling model credits door opening as a backup to room cooling, but like the larger spray systems includes a phenomenological term, because of the difference in geometry.

The ECCS pump room heat loads and room sizes are shown in the Table 2-2 below. This table also summarizes the modeling approach used. The heat loads shown are from the design calculations for LOCA conditions, which tend to be more limiting than normal conditions because of heat transfer from pipes and containment.

TABLE 2-2

ECCS ROOM DESIGN HEAT LOADS AND SIZES

ROOM	HEAT LOAD BTU/HR (POST LOCA)	APPROXIMATE FLOOR AREA (FT^2)	APPROXIMATE ROOM VOLUME (FT^3)	COMMENT
LPCS	416,195	2,072	53,872	Door opening credited as backup cooling method, with additional phenomenological term included to address uncertainty.
RHR A Pump and HX Room	184,277	1,484	38,584	Door opening credited as backup method, based upon similarity to RHR B rooms
RHR B Pump and HX Room	178,400	1564	40,664	Room cooler modeled plus door opening credited based upon CFD analysis.
RHR C	241,162	1,221	31,746	Door opening credited as backup cooling method, with additional phenomenological term included to address uncertainty.
HPCS	570,486	1,722	43,050	Door opening credited as backup cooling method, with additional phenomenological term included to address uncertainty.
RCIC Turbine Room	87,181	1,116	29,016	Door opening credited as backup cooling method, with additional phenomenological term included to address uncertainty.

Switchgear Rooms

The modeling scheme for Switchgear rooms and Switchgear Heat Removal is as follows:

- Switchgear Heat Removal has not been modeled for the Switchgear rooms. GOTHIC analysis for the switchgear rooms has been performed to support continued operability without the availability of the Div 1 or 2 switchgear heat removal systems. Outside air temperatures are assumed to be at the design basis limit. Without any Div. 1 room cooling, room temperatures are shown to be below equipment temperature limits (122°F) for the first 10 hours and marginally over (127.5°F) at 24 hours with no action. Temperatures remain below equipment qualification temperatures for over 24 hours if Div. 1 room doors are opened at 10 hours. Temperatures remain below equipment qualification temperatures for over 24 hours for Div. 2 room with no actions taken. Temporary fan placement would keep room temperatures in specification indefinitely even assuming worst case ambient conditions. As explained previously simple exceedance of EQ limits does not imply immediate equipment failure. Therefore, Switchgear room cooling failures are judged as not being likely failure contributors to switchgear and have not been modeled. Table 2-3 below shows that the Div 1 and 2 switchgear rooms are much larger than ECCS pump rooms even if their heat loads are comparable. Division 3 switchgear room although smaller, also has a lower heat load.

TABLE 2-3

SAFETY RELATED SWITCHGEAR HEAT LOADS AND SIZES

ROOM	HEAT LOAD BTU/HR (NORMAL EQUIP +TRANSMISSION(1))	APPROXIMATE FLOOR AREA (FT^2)	APPROX. ROOM VOLUME (FT^3)	COMMENT
Div 1 Switchgear Room	214,688	3,369	57,273	Not modelled due to slow heatup rate and compensatory actions.
Div 2 Switchgear Room	218,044	5,297	90,049	Not modelled due to slow heatup rate.
Div 3 Switchgear Room	107,885	1,448	23,168	Not modeled.

Note to Table 2-3:

(1) The term Transmission heat load refers to expected heat transfer from adjacent compartments or the outside.

Main Control Room

Cooling for the Main Control Room has not been modeled. See Table 2-4 for its relative size and heat load. Note the loads shown are for LOOP conditions which are slightly higher than those shown for normal operations. The transmission loads shown are nearly half of the total heat load. Since the Main Control Room is at the 800' elevation of the Control Building, which has much of its perimeter exposed to ambient conditions, and minimal perimeter areas with steam or hot piping, most of the transmission heat load would cease as the MCR temperature increases above ambient, reversing the direction of heat transfer. The area shown is large and includes much of what was to have been the Unit 2 side of the MCR, but was never completed. These air volumes are separated by one or two doors, which are part of the same control room envelope and could be left open if the need arose.

The Main Control Room volume has doors that border the outside and can be opened to bring in fresh air, and vent warm Control Room air. There are procedures in place to place gasoline powered fans to improve the air exchange rate with the outside in the event of loss of all MCR cooling such as would occur during an SBO. Because the MCR is permanently occupied, action would be taken to help improve habitability for MCR personnel. These same actions would protect MCR equipment from overheating. Given the relatively large volume relative to heat load and the compensatory actions that the MCR operators could take in the event MCR cooling is lost, not modelling the MCR HVAC system is judged reasonable.

TABLE 2-4

MAIN CONTROL ROOM HEAT LOADS AND SIZES

ROOM	HEAT LOAD BTU/HR (LOOP INCLUDES TRANSMISSION)	APPROXIMATE ROOM FLOOR AREA (FT^2)	APPROX. ROOM VOLUME (FT^3)	COMMENT
Main Control Room	669,483 (includes 300,402 of transmission heat load)	8,360	183,920	Not modelled because of large size relative to heat load. Plus possible compensatory actions exist using temporary fan should the need arise.

RAI 2c

Question

For F&O 1-4, related to supporting requirement (SR) IE-A6, there was no evidence that a systematic evaluation of initiating events due to multiple equipment failures and routine system alignments has been performed. Address whether this systematic evaluation was performed and discuss whether any new initiating events have been identified.

Response to RAI 2c

A systematic evaluation of initiating events was performed for the 2006 PRA model that was peer reviewed, but that systematic evaluation with regards to multiple equipment failures and routine system alignments was not well documented. As part of the 2011 model update, the systematic evaluation was reviewed as part of developing the documentation. No new initiating events were identified.

The Clinton Initiating Events notebook now includes documentation of the systematic review of all systems that have the potential for causing a plant trip. This evaluation includes examination of the impact of multiple equipment failures. For example the Support System Initiating Event fault trees model the multiple equipment failures leading to a trip of the plant. The evaluation of potential initiating events also examines routine system alignments. All systems for which a failure would cause an immediate plant trip are included in the Clinton model as initiating events.

RAI 2d

Question

In F&O 1-14, related to SR IFSN-A6, the peer review team found no evidence that a systematic assessment of the effects of jet impingement, pipe whip, humidity, temperature, etc. on systems, structures and component s (SSCs) was performed. Describe the systematic assessment of these internal flooding effects and summarize its conclusions.

Response to RAI 2d

SR IFSN-A6, as amended by RG 1.200, requires the qualitative assessment of flood-induced mechanisms that are not formally addressed such as jet impingement, pipe whip, humidity, temperature, etc. for Capability Category II. For Capability Category I mechanisms beyond submergence and spray may be omitted from the scope. F&O 1-14 identifies that submergence and spray induced failures were addressed. Therefore, this SR was effectively met for Capability Category I, although the Peer Review report assigned the SR simply as Not Met, rather than Met at CC I.

Protection against the dynamic effects associated with piping rupture is considered to be systematically evaluated as part of the design process. This design process evaluation considers the effect of jet impingement, pipe whip, and environmental concerns such as increased temperature, pressure, humidity, and radiation levels. The internal flooding analysis covers low pressure, low temperature fluids, thus, pipe whip, humidity, and temperature issues are not significant and not modeled as impacts. To supplement the design basis analysis, high energy line break (HELB) related issues are analyzed and modeled separately in the BOC and ISLOCA analysis. The BOC and ISLOCA analysis systematically evaluates the lines connected to the RPV and screens based on a multi-step process. For those lines that are retained for inclusion in the PRA, environmental effects (e.g., harsh environment, temperature, etc.) in the relevant compartments are considered for component failures.

For ISLOCA the following lines are included in the model:

- LPCS Discharge
- RHR LPCI A/B/C Injection
- RHR SDC A and B Discharge
- RHR SDC Suction

For BOC the following lines are included in the model:

- Main steam
- Feedwater
- HPCS
- RCIC
- RWCU

The systematic design basis evaluation of dynamic effects, in conjunction with the systematic evaluation and modeling of BOC and ISLOCA initiators is judged to meet the PRA Standard for SR requirements for CC I which is sufficient for the ILRT application. .

RAI 2e

Question

In F&O 1-17, related to SR IE-C3, it is stated that the basis for recovery actions for certain initiating event fault trees was not documented. Similarly, F&O 1-21, related to SR AS-B3, states that no evaluation for the ECCS pump operation at post-containment venting conditions was provided. The licensee's dispositions of these F&Os indicate that the basis is documented in various plant-specific reports. Provide a discussion of

the basis for crediting the recovery actions mentioned in F&O 1-17 and for crediting ECCS operation after containment venting mentioned in F&O 1-21.

Response to RAI 2e

The following presents the bases for crediting recovery actions associated with WS System and TBCCW System (F&O 1-17) and ECCS operation after containment venting (F&O 1-21).

WS System Recovery Basis & Modeling

Recovery events for the WS System are documented in the WS System Notebook, which provides the following information:

> A recovery failure probability of 1.0 is assumed for scenarios with "start" failures or heat exchanger failures. A recovery failure probability of 0.50 is credited for scenarios with "run" failures; this estimate is judged reasonable given generic industry experience (i.e., NSAC-161, Faulted Systems Recovery Experience, estimates a non-recovery probability of ~0.50 at two hrs after failure of an "Other-Non-Safety System"). A value of 0.10 is used for plugged strainers given that recovering a plugged strainer is judged to be more straightforward and easier to perform (e.g., manual back washing, bypass strainers), and strainer plugging is a slowly developing failure mode.

There are also two strainers on the discharge header, both normally in service, but either of which can supply loads required to prevent core damage or containment failure. There is also a bypass around the strainers that is not currently modeled but can be credited as a recovery of failed or plugged strainers.

The pump run failure recovery event does not appear in initiating event tree cutsets at a truncation level of 1E-6 /yr. Therefore, if the recovery failure probability of 0.50 is increased to 1.00, there would be a negligible impact to the WS system initiating event frequency. It should be noted that success criteria is 1 of 3 WS pumps running.

The WS IE has a Fussell-Vesley (FV) of 1.54E-03 or ~0.15% (Ref. Table 3.2-4 of the CPS PSA-013 Clinton PRA Summary Notebook, Rev. 4, March 2014). Although recovery probabilities are reasonable, changes to the probabilities would have an insignificant impact to CDF.

TBCCW Recovery Basis & Modeling

Recovery events for the TBCCW System are documented in the TBCCW System Notebook, which provides the following information:

> A recovery failure probability of 1.0 is assumed for scenarios with "start" failures (which include backflow failures) or heat exchanger failures. A recovery failure probability of 0.50 is credited for scenarios with "run" failures; this estimate is judged reasonable given generic industry experience (i.e., NSAC-161, Faulted Systems Recovery Experience, estimates a non-recovery probability of ~0.50 at two hrs. after failure of an "Other-Non-Safety System").

The TBCCW initiating event frequency is 6.03E-03/yr with the IE fault tree run at 1E-07/yr truncation. Failure to recover a pump train given initial pump failure basic event was set to 1.0 (from 0.50) and the IE frequency recalculated. There was no increase in IE frequency at the 1E-07/yr truncation limit.

A second recovery basic event is used in the IE fault tree to credit recovery of a significant TBCCW system leak. This event has a probability of 0.10, based on the likelihood the system leakage is trended (small leakage precursor identified) and there is a high likelihood the leak would be isolated should leakage increase significantly. Given lack of plant experience a quantitative assessment is provided. Information Notice No. 98-25 "Loss of Inventory from Safety-Related, Closed-Loop Cooling Water Systems" discusses three events of loss or potential loss of inventory due to leakage. Only one leak event (at Palisades) caused a loss of the system. The leak was isolated and the system returned to service in 15 minutes. Industry data supports a low failure to recover probability for leakage events.

Setting this event to 1.0 (from 0.10) results in a TBCCW IE frequency increase of 9.63E-03/yr. An increase of approximately 50%. The TBCCW IE has a Fussell-Vesley (FV) of 1.27E-03 or ~0.13% (Ref. Table 3.2-4 of the CPS PSA-013 Clinton PRA Summary Notebook, Rev. 4, March 2014). This indicates that an increase of the IE frequency by 50% would increase CDF by less than 0.1%.

WS and TBCCW Recovery Event Impact to the CPS ILRT LAR Risk Assessment

Given the low importance of the WS and TBCCW initiating events, the probabilities used for recovery would not impact the ILRT LAR overall conclusions.

ECCS Operation Post Containment Venting (F&O 1-21)

CPS Component Data Notebook Appendix G.13 provides justification for a high probability that ECCS pumps would have adequate NPSH during and following containment venting. Appendix G.13 summary statements include the following:

> "Based on consideration of the available vent paths, containment depressurization during vent operations is expected to be relatively slow. This coupled with the deterministic CPS calculations indicate that the suppression pool depth will provide adequate overpressure to prevent the

suction pipe from flashing to steam even though the surface of the suppression pool may be in bulk boiling.

Given these calculations, it is judged likely that the vent operation would not be so rapid that vapor locking or steam binding in the ECCS pump suction would occur. In addition, the ECCS suction lines are equipped with drilled holes in the tops of the inverted-U piping that minimize the potential for steam binding.

Therefore, the phenomenological common cause event that fails all ECCS pumps with suction from the pool due to steam binding following failure to control containment venting [1SY--STEAMBOUND-] is assigned a likelihood probability of 1E-2."

Appendix G.13 contains additional detail and is partially reproduced below for the remainder of the response.

G.13.2 Containment Vent Impact on ECCS Continued Operation (1SY--STEAMBOUND-)

Containment venting is one of multiple methods of controlling containment pressure. However, the success of containment venting has a potential to create adverse conditions on the suction of pumps aligned to the suppression pool. It is postulated that rapid depressurization of the containment by opening a large vent line would create conditions of bulk boiling in the suppression pool and attached piping. This could lead to steam binding of the suction piping due to the collection of steam in the inverted "U" suction piping for all of the ECCS pumps.

The assessment of the ability to maintain ECCS following vent initiation includes an evaluation of the following:

- Physical arrangement
- Vent lines
- Procedural guidance
- Training
- Available calculations

Physical Arrangement

The physical arrangements that are considered important are the following:

- The depth of water above the pump suction
- The physical configuration of the suction pipe
- The size of vent lines

This physical arrangement could be postulated to result in steam vapor collecting in the inverted "U" if a rapid pressure change occurred in the pipe, e.g., containment pressure is reduced from PCPL (~45 psig) to 0 psig with the water initially at 293°F. However, two issues make this unlikely:

1. CPS MAAP runs show that if the containment vent is opened at the 45 psig vent pressure and left completely open (no control), the containment depressurization rate is slow (6 hrs. to reduce containment pressure from 45 psig to 35 psig).
2. The tops of each ECCS suction line coming out of the suppression pool contain 64 3/32" diameter drilled holes. These holes would minimize the potential for steam binding in the inverted-U configuration.

Vent Lines (see Procedure 4411.06)

There are a large number of containment vent lines available as options for venting the containment.

Emergency Containment Venting (CPS No. 4411.06) includes 3 vent paths that are equivalent or greater in size to an orifice of six inches in diameter. With this criteria, any one of the three vent paths is capable of venting the containment. The three vent paths modeled in the PRA are:

1. Venting the Containment to the Spent Fuel Pool via the Residual Heat Removal A (RHR A) and Fuel Pool Cooling and Cleanup Systems
2. Venting the Containment to the Spent Fuel Pool via the Fuel Pool Cooling and Cleanup (FC) System containment return header, and
3. Venting directly to the atmosphere via Continuous Containment Purge (CCP).

In addition to the three vent paths with equivalent diameters > 6", there are three other vent paths that have smaller equivalent diameters that would allow for a more controlled venting.

Procedural Guidance

The EOPs provide direction for Primary Containment Control (4402.01):

- Vent to stay below PCPL (~ 45 psig)

This implies that pressure would likely be controlled to maintain a pressure band below 45 psig.

The TSC would likely be operational at this time (> 24 hours into an event where containment parameters are far outside of the EOP entry conditions). The TSC would use the Technical Support Guidelines (CPS 4750.01) which provide additional insights and decision making guidance for accidents beyond the design basis.

- TSG Tab 6 (p. 42 of 83) requires the team to select a control band for venting. The criteria for selecting the control band include the following:
  - Assure adequate RPV injection
  - Prevent loss of suction for RPV injection or sprays
  - Minimize loss of non-condensibles

This TSG information leads to the conclusion for the loss of DHR sequences that the TSC would specify a control band on the venting and would not arbitrarily authorize rapidly depressurizing the containment. In addition, TSG Tab 6 also includes the following:

- P. 33 of 83 of 4750.01 specifies selecting the smallest vent path that is adequate
- P. 33 and 35 of 83 of 4750.01 also identify the potential for adverse impacts on plant equipment and personnel. These provide additional significant incentive to minimize any venting to protect:
  - Personnel safety
  - Equipment operability

On the other hand, in discussions with the EOP coordinator, he indicated that there is no explicit procedural guidance to state that loss of NPSH is likely given a rapid containment depressurization.

Training Guidance

The EOP training which includes an understanding of the EOP bases provides guidance on the use of the vent paths:

- Select the vent with the best control that meets the needs for maintaining pressure below PCPL
- Minimize the release of radionuclides
- Venting should not be performed indiscriminately
- Controlled releases should be performed in a manner that minimizes the total dose to the public

This training also would likely result in a control band near 45 psig.

Table G.13.2-1 summarizes the calculated NPSH for the cases evaluated including 212°F in the suppression pool for a post LOCA draw down. All calculations at this temperature indicate significant margin to the 5 ft. of NPSH required for ECCS pump operation. Therefore, the ECCS pumps are not jeopardized by loss of NPSH for the most limiting of design basis accidents.

Table G.13.2-1

NPSH FOR ECCS PUMPS

PUMP	NPSH[1] GIVEN SUPPRESSION POOL CONDITION		
	212°F	212°F WITH DEBRIS	185°F WITH DEBRIS
LPCS	12.9	12.8	27.8
HPCS	12.4	12.3	27.4
LPCI A	13.3	13.2	28.3
LPCI B	13.6	13.6	28.6
LPCI C	14.2	14.2	29.2

Note to Table G.13.2-1:

(1) Requirement ~ 5 ft. NPSH minimum

Alternatives

HPCS operation may be provided with suction from either the suppression pool or the RCIC tank. The use of the RCIC tank would preclude concern over the impact of venting on continued HPCS operation. However, the RCIC tank has limited volume that is not expected to be sufficient for HPCS operation over the required time frame. This would require a separate crew action to refill the RCIC tank to support the extended HPCS operation. This crew action is treated separately and is included in the Clinton HRA Notebook.

ECCS Post Containment Venting Summary

Based on consideration of the available vent paths, containment depressurization during vent operations is expected to be relatively slow. This coupled with the deterministic CPS calculations indicate that the suppression pool depth will provide adequate overpressure to prevent the suction pipe from flashing to steam even though the surface of the suppression pool may be in bulk boiling.

Given these calculations, it is judged likely that the vent operation would not be so rapid that vapor locking or steam binding in the ECCS pump suction would occur. In addition, the ECCS suction lines are equipped with drilled holes in the tops of the inverted-U piping that minimize the potential for steam binding.

Therefore, the phenomenological common cause event that fails all ECCS pumps with suction from the pool due to steam binding following failure to control containment venting [1SY--STEAMBOUND-] is assigned a likelihood probability of 1E-2.

RAI 2f

Question

For F&O 1-24, which is related to SR SY-A18, identifies the lack of a concerted effort to identify accident conditions that could cause system failures. F&O 1-22, related to SR AS-B3, also raises a similar concern. Briefly describe the process followed to identify, review, and model accident conditions that could cause system failures.

Response to RAI 2f

For the 2011 PRA update each system notebook was reviewed as part of the update process to ensure that the documentation accurately depicts the PRA model. As part of this update review of accident conditions that could cause system failures were checked. These aspects are documented in the system notebooks and consider the following:

- Component location in the plant (Section 2.4)
- Spatial dependencies (Section 2.5)
  - Credit given accident conditions in the containment
  - Credit for containment failure or vent conditions
  - Whether the system is a HELB or ISLOCA source
  - Room cooling requirements (detailed in the Dependency Notebook)
  - Internal flooding aspects (detailed in the Internal Flood Notebook)

- Performance during accident conditions (Section 4)
  - Success criteria
  - Initiator impact on the system
  - Impact of the system on other systems.

Interviews with system engineers were also conducted to ensure that system modeling is consistent with the as-built, as-operated plant. Interviews included the following:

- Review of key modeling assumptions
- Review of support system dependencies assumed in the model
- Review of system success criteria
- Discussion of operational experience history
- Discussion of potential system failures leading to plant initiating events
- Discussion of equipment operation beyond design limits

Based on the above process and actions, the PRA model and documentation was enhanced to ensure that accident conditions that could cause system failures are appropriately included.

RAI 2g

Question

For F&O 1-27, which is related to SR SY-A24 and DA-C15, it was found that plant-specific data was not analyzed to support crediting RHR repair in the probabilistic risk assessment (PRA) model. The resolution to this F&O states that the repair data are based on industry experience as allowed by SR DA-C15 and are judged to be reasonable. The NRC staff's endorsement of the 2009 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009 in Regulatory Guide (RG) 1.200, Revision 2, includes the exception for SR DA-C15, which states that industry experience should be used only if plant-specific experience is insufficient to estimate the failure to repair. Explain why the plant specific experience is insufficient to estimate the failure to repair.

Response to RAI 2g

Plant specific data for RHR recovery is judged to be insufficient to support a recovery estimate due to few occurrences (only two in the last five years). The two occurrences generally support the probability estimate from generic data.

The RHR recovery term used in the Clinton PRA model based on industry data (1RHRXDHRRECLTH-- FAILURE TO RECOVER DECAY HEAT REMOVAL LONG TERM) has a conservative failure probability of 0.42 and is based upon a Mean Time To Repair Model (MTTR), with a recovery time available of 23.6 hours. This available recovery time is based upon not exceeding the Primary Containment Pressure Limit (PCPL) for transient cases with no containment heat removal. A second RHR recovery term (1RHRX-REC-UPDH-- RHR FAILURE TO RECOVER WITH UPPER POOL DUMP SUCCESS) has a failure probability of 0.27 and is based on a recovery time of 39.8 hours.

A review of the Clinton maintenance rule records for RHR was performed covering approximately a 5 year period since 2012 identified two (2) functional failures during this time.

1. In September of 2014 the handswitch for suppression pool return valve 1E12F024B failed. 29.7 hours of unavailability was accrued during the time it took to repair the handswitch in non-emergency conditions. The valve is used to return water flow to the suppression cool when utilizing RHR B in suppression pool cooling mode. However, if this loop of suppression pool cooling were needed during a plant event, this particular valve, which is assessable outside of containment, could have been readily opened by operators using its valve handwheel.
2. In December 2016 the RHR C pump 1E12C002C failed to start during a pump surveillance. This was caused by misalignment of components on the pumps 4.16kv breaker. Unavailability time accrued to make the needed repairs was 10.3 hours.

The plant specific experience associated with RHR recovery is sparse but is considered adequately represented by the relatively modest non-recovery factor developed from industry data.

RAI 2h

Question

For F&O 1-43, which is related to SR QU-D1, it is stated that a review of the "significant" cutsets and accident sequences as defined in the standard was not performed. The resolution states that the top cutsets and accident sequences now represent a larger percentage of the total core damage frequency (CDF) compared to the time of the peer-review, but does not address the concern in the F&O. Similarly, F&O 1-46, related to SR QU-D5, states that a review of the non-significant cutsets and accident sequences was not performed. The resolution for F&O 1-46 contains a reference to a licensee guidance document, but does not address the concern of the F&O. Address whether a review of significant cutsets and accident sequences, as defined in the 2009 ASME/ANS PRA Standard RA-Sa-2009, as well as non-significant cutsets and accident sequences was performed.

Response to RAI 2h

A review of significant and nonsignificant cutsets and accident sequences was performed for the 2011 and 2014 PRA model updates in accordance with the 2009 ASME/ANS PRA Standard RA-Sa-2009. Since the ILRT risk assessment is based upon the 2014 PRA model, the focus of this response is upon that model.

The pertinent PRA Standard Support Requirements specify the following:

- QU-D1 REVIEW a sample of the significant accident sequences / cutsets sufficient to determine that the logic of the cutset or sequence is correct.
- QU-D5 REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.

For the 2014 PRA model, a review of the top ten CDF and LERF cutsets and accident sequences is explicitly documented in the PRA Summary Notebook in which the cutsets and accident sequences are listed and explained. These cutsets compose 20% of CDF and 16% of LERF, and the accident sequences compose 82% and 93% of CDF and LERF, respectively. This explicitly documented review of cutsets and sequences is one important means of reviewing a sampling of significant cutsets and accident sequences. This focus of this review is to ensure the top cutsets and sequences exhibit correct logic and are reasonable. Consideration is given as to how the top cutsets and accident sequences differ from the previous update to ensure that changes are appropriate.

In addition to the explicit listing, significant cutsets and accident sequences are also reviewed as part of satisfying other PRA Standard SRs associated with model quantification. These would include:

- QU-D2 – Reviewing results for model consistency
- QU-D3 – Reviewing results to determine that flag settings, mutually exclusive rules, and recovery rules yield logical sense
- QU-D4 – Comparing results to similar plants
- QU-D6 – Identifying significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors
- QU-D7 – Reviewing importance of components and basic events.

As part of these reviews for other SRs, significant and nonsignificant cutsets and accident sequences are reviewed.

During the model update development process, interim quantifications are also performed to examine the impact to the model for specific model changes. For each of the changes, cutsets and accident sequence results are generally reviewed to confirm that the impacts of the model changes are correct and understood. These interim quantifications would involve both significant and nonsignificant cutsets and accident sequences.

For the 2011 CPS PRA update, an Exelon internal "challenge review" was conducted just prior to model finalization in which senior PRA engineers who are independent from the PRA update team spend about 2 days performing an independent review of and comparison of model results. During this review the reviewers examine the various technical areas of the model, model changes, and reviews significant and nonsignificant cutsets and accident sequences.

In summary, a review of significant and nonsignificant cutsets and accident sequences was performed for the 2011 PRA model update in accordance with the 2009 ASME/ANS PRA Standard RA-S1-2009 via:

- Explicitly documented reviews in the PRA Summary Notebook
- Reviews conducted in support of other QU SRs
- Model development interim results reviews
- An independent internal "challenge review"

These reviews accomplished a thorough sampling of cutsets and accident sequences as specified by SR QU-D1 and SR QU-D5.

RAI 3

Question

In the March 31, 2006 supplement, resolution of F&O 5-7 states:

> The [Electric Power Research Institute] EPRI [Human Reliability Analysis] HRA Calculator is now used to quantify the probabilities of the [Human Error Probabilities] HEPs in the Clinton model...The transition to using the EPRI HRA Calculator did not represent a methodology change...

In addition, resolutions of all F&Os related to the human reliability (HR) supporting requirements (F&Os 1-31, 1-32, 1-33, 1-34, 2-16, 3-13, and 5-10) seem to indicate that many HRA related changes to the PRA model were performed in the 2011 PRA update and/or as a result of using the EPRI HRA Calculator.

The 2009 ASME/ANS PRA Standard RA-Sa-2009 defines a PRA upgrade as,

> ...incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences.

Non-mandatory Appendix 1-A of the PRA Standard cites "a different HRA approach to human error analysis..." as a potential PRA upgrade. Based on Section 1-5, "PRA Configuration Control," of the standard and RG 1.200, Revision 2, Regulatory Position 1.4, "PRA Development, Maintenance, and Upgrade," a PRA upgrade must be peer reviewed.

a. Justify why this model change is not considered a PRA upgrade requiring a focused-scope peer review. If this change qualifies as an upgrade, provide the results from of the focused-scope peer review addressing the associated F&Os and their resolutions.
b. Provide an overview of all other changes in the internal events PRA model that occurred after the 2009 peer review, and clarify whether any of these changes qualify as a PRA upgrade that would require a focused-scope peer review.

Response to RAI 3a

For the 2011 Clinton FPIE, the update HRA was entered into the EPRI HRA Calculator. The use of the HRA Calculator is to ensure consistency among the HEPs and to enhance documentation of the HEPs. The methods (e.g., ASEP, THERP, CBDT) used for calculating the Clinton HEPs was unchanged from the Peer Review. The use of the HRA Calculator tool is consistent with a PRA update. Similar to Example 11 of Appendix 1-A changing software while maintaining similar methodology is not considered a PRA upgrade. Example 11 notes the importance of comparing differences between the old and new software results. This comparison was performed during the 2011 HRA update into the HRA Calculator.

With regards to the other HRA related F&Os noted in this RAI, the following additional information is provided:

- F&O 1-31 (HEP Consistency Check) – The F&O notes that HEP values were compared against those of previous models, but there was no documentation of checks of consistency between individual HFEs in the peer review documentation. This lack of documentation was remedied by adding Appendix D in the 2011 HRA notebook. Documentation enhancements represent PRA Maintenance.

- F&O 1-32 (Screening Values for Dependent HFE Identification) – The F&O notes that the screening values used for identifying dependent HFE combinations may have been insufficient (i.e., too low). For the 2011 PRA update, post-initiator HEPs were set to 0.1 or higher to ensure relevant dependent HFEs were properly identified. Additional dependent HFEs were included in the 2011 PRA as a result. The addition of these new dependent HFEs did not have a significant impact on the risk insights. Including these new HFEs is similar to Example 20 of Appendix 1-A of the PRA Standard. Additionally, all independent HEPs were set to their correct values after applying the dependent HEPs. Updating the screening values for the HEPs represents PRA Maintenance.
- F&O 1-33 (Nominal Values for Dependent HFE Identification) – This F&O is similar to F&O 1-32 in that some HFEs inadvertently retained their nominal values during dependency assessment. This F&O was addressed in conjunction with F&O 1-32 as part of updating the HRA dependency analysis for the 2011 PRA update. As discussed above for F&O 1-32, these changes represent PRA Maintenance.
- F&O 1-34 (Nominal Values for Dependent HFE Identification) – This F&O is essentially redundant to F&O 1-33 and similar to F&O 1-32 in that some HFEs inadvertently retained their nominal values during dependency assessment. When the Peer Review team ran their own dependency assessment with elevated HEPs, they identified additional combinations. This F&O was addressed in conjunction with F&O 1-32 as part of updating the HRA dependency analysis for the 2011 PRA update. As discussed above for F&O 1-32, these changes represent PRA Maintenance.
- F&O 2-16 (Pre-Initiator Identification for Multiple Impacts) – This F&O identifies a lack of documentation associated with identification of pre-initiator HFEs that may impact multiple trains or redundant / diverse SSCs. For the 2011 PRA update the documentation was enhanced to discuss the systematic approached used to identify pre-initiators. This change represents a PRA Maintenance.
- F&O 3-13 (Pre-Initiator Identification Inadequate) – This F&O identifies that while risk significant pre-initiators were identified, other pre-initiators which were expected to be of low risk significance were not pursued. An additional 16 pre-initiators were developed for the 2011 model consistent with the approached used in the previous PRA update to bring the total to 107. For the Clinton model the pre-initiators are of low risk significance for both CDF and LERF. This change represents PRA Maintenance.
- F&O 5-10 (Documentation Insufficient for HFE Attributes) – This F&O identifies lack of documentation evidence for certain HFE attributes. Use of the HRA Calculator in the 2011 update addressed this documentation issue. As discussed above, use of the HRA Calculator is determined to be PRA Maintenance.

Additionally, an independent review of the HRA was performed the week of February 20th, 2017 by an experienced HRA analyst[1] to confirm that none of the changes made to the HRA after the peer review would qualify as PRA upgrade. Several areas of interest were noted:

1. There was a wider use of Cause Based Decision Tree Method (CBDTM) in the 2011 HRA update coincident with implementation of the HRA Calculator. The 2009 peer review report noted the use of the CBDTM under supporting requirement (SR) HR-G3 and noted the use of CBDTM for diagnosis errors. The peer review team reviewed the use of CBDTM and found it to be acceptable. Therefore, it is judged that this is not a new methodology, rather an expanded use of the existing method. (This is similar to Example 21 of Appendix A-1 of the PRA Standard.)
2. A review of the pre-initiator HEPs developed for the peer review model and the subsequent 2011 model (i.e., before and after using the HRA Calculator) found that the same ASEP method was used and that, in general, the HEPs are consistent and within the error factor for the HEPs given. The pre-initiator review found that some HFEs were determined to be no longer needed in the 2011 model. These HFEs were conservatively set to a low value (1E-06) in the model. This was not an upgrade because there was no change in methodology. Additionally, while there was a change in capability category, it was judged that no significant accident sequences were impacted. Therefore, it is not an upgrade.
3. The HRA SRs related to the F&Os were reviewed and assessed for technical adequacy (i.e., capability category). The review found that the 2014 Clinton model meets or exceeds Capability Category II for all related HRA SRs. The following table shows the results from the independent review.

[1] The analyst has approximately 27 years in the nuclear industry, with 10 years as a PRA analyst and specifically as an HRA analyst. The analyst has been involved in several full power internal events PRA peer reviews and has detailed knowledge of the PRA Standard. The review was independent in that the analyst neither worked on 2016 CPS ILRT nor the CPS HRA in the 2006C, 2011 or 2014 updates.

SR	2009 PEER REVIEW ASSESSMENT	2017 INDEPENDENT REVIEW ASSESSMENT	COMMENT
HR-A1	SR Not Met	SR Met	F&O properly addressed
HR-A2	SR Not Met	SR Met	F&O properly addressed
HR-A3	SR Not Met	SR Met	F&O properly addressed
HR-B1	SR Met (CC II/III)	SR Met (CC II/III)	No change in capability
HR-B2	SR Not Met	SR Met	F&O properly addressed
HR-C1	SR Met	SR Met	No change in capability
HR-C2	SR Met (CC I)	SR Met (CC II/III)	F&O properly addressed
HR-C3	SR Met	SR Met	No change in capability
HR-D1	SR Met	SR Met	No change in capability
HR-D2	SR Met (CC I)	SR Met (CC II/III)	F&O properly addressed
HR-D3	SR Met (CC I)	SR Met (CC II/III)	F&O properly addressed
HR-D4	SR Not Met	SR Met	F&O properly addressed
HR-D5	SR Not Met	SR Met	F&O properly addressed
HR-D6	SR Not Met	SR Met	F&O properly addressed
HR-D7	SR Met (CC I/II)	SR Met (CC III)	F&O properly addressed
HR-E1	SR Met	SR Met	No change in capability
HR-E2	SR Met	SR Met	No change in capability
HR-E3	SR Met (CC II/III)	SR Met (CC II/III)	No change in capability
HR-E4	SR Met (CC II/III)	SR Met (CC II/III)	No change in capability
HR-F1	SR Met (CC I/II)	SR Met (CC I/II)	No change in capability
HR-F2	SR Met (CC II)	SR Met (CC II)	No change in capability
HR-G1	SR Met (CC II)	SR Met (CC II)	No change in capability
HR-G2	SR Met	SR Met	No change in capability
HR-G3	SR Met (CC II/III)	SR Met (CC II/III)	No change in capability
HR-G4	SR Met (CC II)	SR Met (CC II)	No change in capability
HR-G5	SR Met (CC II)	SR Met (CC II)	No change in capability
HR-G6	SR Not Met	SR Met	F&O properly addressed
HR-G7	SR Not Met	SR Met	F&O properly addressed
HR-G8	SR Met (CC II)	SR Met (CC II)	No change in capability
HR-H1	SR Met (CC II)	N/A	Not reviewed
HR-H2	SR Met	N/A	Not reviewed
HR-H3	SR Not Met	SR Met	F&O properly addressed
HR-I1	SR Met	N/A	Not reviewed
HR-I2	SR Met	N/A	Not reviewed
HR-I3	SR Not Met	N/A	Not reviewed
QU-C1	SR Not Met	SR Met	F&O properly addressed
QU-C2	SR Not Met	SR Met	F&O properly addressed

Based on the above discussions for use of the HRC Calculator and resolution of individual HRA related F&Os, the changes made are determined to be PRA Maintenance.

Response to RAI 3b

There have been two updates to the internal events PRA model since the performance of the 2009 peer review of the 2006 (i.e., CL06C) PRA model. The CDF and LERF values from the peer reviewed model and models of record since the peer review are provided below.

MODELS	CDF	LERF
CL06C (2009 Peer Review Model)	5.57E-06(1)	1.20E-07(2)
CL11A	3.16E-06(2)	1.23E-07(3)
CL14A	2.13E-06(3)	1.16E-07(4)

A summary of changes made to the internal events PRA models of record are documented in the introduction sections of the respective FPIE PRA Update Quantification Summary Notebooks (CPS-PSA-013). Additionally, changes in the FPIE PRA models are tracked in the Clinton Updating Requirements Evaluation (URE) database which tracks PRA observations and open items identified in between scheduled FPIE PRA Update periods. URE items that involve model changes that were addressed in the updated models are also listed and described in the Quantification Summary Notebooks. The summary of changes and table of UREs involving model changes were used to identify the changes made since the CL06C model was peer reviewed.

A summary of the identified changes for the 2011 and 2014 PRA model are discussed below. For each change, a discussion is also provided to identify whether the change item is *PRA Maintenance* or *PRA Upgrade*. A reference to the specific PRA Standard Appendix 1-A "Example" in which the change item best relates is also provided.

These changes, including any new analyses or incorporation of new methodologies performed in the internal events PRA model since the last full-scope peer review from 2009, were reviewed for this RAI response. None of the changes made during the 2011 or 2014 PRA model updates meet the definition and criteria of ASME/ANS RA-Sa-2009 for a PRA upgrade. The following list includes the PRA model changes and a discussion of why they are considered PRA maintenance:

(1) At truncation of 1E-11/yr using the single top PRA model.

(2) At truncation of 1E-12/yr using the single top PRA model.

(3) At truncation of 5E-13/yr using the single top PRA model.

(4) At truncation of 5E-14/yr using the single top PRA model.

CL11A Model Changes

The CL11A model was developed as a result of a regular scheduled update, with a focus on resolving Peer Review F&Os. Major changes incorporated into the model include the following data, plant, procedure, and analysis changes:

1. Bayesian updated initiating event frequencies utilizing the most recent Clinton operating experience and the most current generic data.
   - ***PRA Maintenance*** (Examples 2 and 3). Using new plant-specific and new generic data, no new methodology employed. This was not the first time Bayesian updating was performed for such data.
2. Revised component failure data including use of most recent plant-specific component failure data gathered from the site.
   - ***PRA Maintenance*** (Examples 2 and 3). Using new plant-specific and new generic data, no new methodology employed.
3. Bayesian updated individual component random failure probabilities using plant-specific data (where applicable) and the most current generic sources.
   - ***PRA Maintenance*** (Examples 2 and 3). Although this was the first Bayesian Update for random failure probabilities used in Component Data, the same Bayesian Update methodology that was used for Initiating Event frequencies was employed. Therefore, this would not be a new methodology just an expansion of current methodology to a different data set.
4. Removal of Division 3 diesel crosstie procedure
   - ***PRA Maintenance*** (Examples 6 and 7). Logic model enhancement, no new methodology employed compared to the prior model. (This was re-instated in the 2014 model update)
5. Update of HRA calculation with EPRI HRA Calculator® software. There is no change in the HRA methods used.
   - ***PRA Maintenance*** (Examples 20 and 11). HEP modeling enhancement, no new methodology employed. This item is discussed in more detail in response to RAI 3a.
6. Edits to Clinton Internal Flood analysis
   - ***PRA Maintenance*** (Examples 6 and 7). Logic model enhancement, no new methodology employed compared to the prior model.
7. Maintenance unavailability data based on the most recent plant operating experience.
   - ***PRA Maintenance*** (Examples 2 and 19). Using new plant-specific data, no new methodology employed.

8. Changes associated with new RAT alignment. CPS now has 3 RATs. System Logic amended to add UA and UR for 3 RATs and new CB.
   - ***PRA Maintenance*** (Examples 6 & 7). Logic model enhancement, no new methodology employed compared to the prior model.
9. URE CL2008-007 - TBCCW Train in standby basic events replaced with separate standby basic events for each pump and heat exchanger.
   - ***PRA Maintenance*** (Examples 6 & 7). Logic model enhancement, no new methodology employed compared to the prior model.
10. URE CL2010-006 - SX manual alignment probabilities changed due to the increased amount of time required to complete the action.
    - ***PRA Maintenance*** (Examples 6 & 20). HEP modeling enhancement, no new methodology employed.
11. URE CL2010-002 - Success of HPCS room cooling using 1 of 2 Room Coolers during cold winter months.
    - ***PRA Maintenance*** (Examples 6 & 7). Logic model enhancement, no new methodology employed compared to the prior model.
12. Changed number of running condensate pumps from 4 of 4 to 3 of 4 per system manager interview.
    - ***PRA Maintenance*** (Examples 6 & 9). Logic model enhancement based on new knowledge, no new methodology employed compared to the prior model.
13. Addition of plant availability factor into the model logic.
    - ***PRA Maintenance*** (Examples 6 & 7). Logic model enhancement, no new methodology employed compared to the prior model.
14. Existing Support System Initiating Event Fault Trees for TBCCW, CCW, and WS incorporated in the single-top model.
    - ***PRA Maintenance*** (Example 5). Prior to incorporating the support system initiating event (SSIE) fault trees into the single-top model, they were stand-alone fault trees that were quantified to provide the point estimate input into the main fault tree model. The impact of directly incorporating the SSIE fault trees into the main model was small (~0.3% impact on CDF, ~0.1% impact on LERF). Had new fault trees been developed to replace point estimates (e.g., from generic data), this could constitute a PRA Update based on Example 5. However, since plant-specific fault trees had been previously developed and peer reviewed, their simple incorporation into the main model is judged PRA Maintenance. The sensitivity case performed at that time confirmed that there was not a significant impact on quantification results or PRA insights.

15. URE 2010-057 - Update of coincident maintenance events with plant specific data.
    - ***PRA Maintenance*** (Examples 2 and 19). Using new plant-specific data, no new methodology employed.
16. Removed RCIC water leg pump from model.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.
17. Updated Main Control Room Abandonment probability for Internal Flood
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model. Minor impact to CDF of 4% and LERF of 1%.
18. Added digital FW failure to prevent high level event to RPV Level Control gate.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.
19. Additions to the MEX file regarding LOCA locations.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.
20. URE CL2010-007 - Modifications to fault tree logic for removal of QS relay to prevent trip to apply only during ATWS scenarios. Current procedures only direct this action during ATWS scenarios.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.
21. Incorporation of RHR A/B room cooling calculation results and update of door opening HEP probability.
    - ***PRA Maintenance*** (Examples 6, 7, 9 & 20). Logic model enhancement based on new knowledge, no new methodology employed compared to the prior model. HEP modeling enhancement.
22. URE CL2010-065 - Removed ISLOCA early isolation credit for closing MOVs.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.
23. Use of Alpha factors for Common Cause Failures.
    - ***PRA Maintenance*** (Example 3 and 26). The change from MGL to Alpha factor method (NUREG/CR-5497) reflects Example 26 because MGL and Alpha methods both correlate to the same result. This change is less extensive than a method change from the Beta Factor approach of Example 27.

24. Added credit for automatic initiation of containment spray system.
    - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.

CL14A Model Changes

25. Updated model to reflect removal of ADS Inhibit step from EOPs for non-ATWS Scenarios
    - ***PRA Maintenance*** (Example 6, 22 and 20). Logic model enhancement due to revised procedure, no new methodology employed compared to prior model.
26. Credited revision of Division 3 diesel cross-tie procedure
    - ***PRA Maintenance*** (Example 20 and 22). Logic model enhancement due to revised procedure, no new methodology employed compared to prior model.

RAI 4

Question

One of the topical report use conditions identified in the safety evaluation (SE) approving the methodology in EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105) is that the average leak rate for the pre-existing containment large leak class (i.e., Class 3b) be assigned a value of 100 times the maximum allowable containment leakage rate (L_a) instead of the previously used value of 35 La. Due to the methodology used, the pre-existing drywell large leak also uses the new postulated leakage rate (i.e., 100 times the maximum allowable drywell leakage rate [DWLb]). These leakage rates (i.e., 100 La and 100 DWLb) represent a substantial increase from the values used for the previous one-time frequency extension request (ADAMS Accession No. ML030370524).

In Section 4.1.3 of Attachment 4 to the submittal, the licensee refers to the drywell bypass leakage rate test (DWBT) risk assessment methodology used for its previous one-time ILRT/DWBT extension request as the basis for the approach used. In the same section, the licensee discusses the impact of drywell leakage on containment over-pressurization and CDF based on the margin found in the results of the deterministic calculations documented in the previous one-time ILRT/DWBT frequency extension request. However, discussion of the impact of the increased DBWT leakage of 100 DWLb on the amount of Cesium Iodide (CsI) released is not provided. Therefore, address why the increased DWBT leakage of 100 DWLb does not change the assignment to the EPRI accident Class 3a and Class 3b shown in Table 4.6-1 of Attachment 4 to the submittal.

Response to RAI 4

Table 4.6-1 from Attachment 4 of the submittal is reproduced below. The focus of the response is on DWBT leakage of 100 DWL_b. Specifically why the increased DW bypass leakage of 100 DWL_b does not change the assignment to the EPRI accident Class 3a and Class 3b. The leakage combinations with DW Bypass leakage at 100 DWL_b are CA'1, CB'1 and CC'1 as shown below.

The change from 35 DWL_b to 100 DWL_b will not impact leakage combinations already classified as 3b. The assignments for CB'1 and CC'1 are EPRI Class 3b. Class 3b leakage is assumed to be LERF contributor. Therefore, the only leakage combination requiring further analysis is CA'1.

The CA'1 3a assignment is conservative for the following reasons:

- Containment leakage (to the environment) is limited to Tech Spec allowed leakage (1La).
- Drywell Bypass leakage DWLb at 300 scfm is an order of magnitude lower than the Drywell Bypass leakage tech spec limit of 3654 scfm (10 DWLb = 3000 scfm).

Thus, a qualitative assessment of a scenario with drywell leakage at ~ 10 DWL_b and containment leakage at 1 L_a finds that the release would be far below a Large release.

Additionally, a MAAP case performed in support of the one-time deferral of the primary containment Type A test bounds the leakage combination CA'1 and provides additional support for not classifying this scenario as EPRI Class 3b. The drywell leakage is 156 DWL_b which is greater than the 100 DWL_b used in the present methodology. The inputs and results of this MAAP run are the following:

- MAAP Calculation: CL0010
- Scenario: Design Basis LOCA, No wetwell sprays, No injection
- Drywell Leakage: 156 DWL_b (1.18 sq. ft. hole size)
- Containment Leakage: 1 L_A (0.000118 sq. ft. hole size)
- Fractional Cesium Iodide Release to Environment: 6.1E-5

The threshold developed for LERF is generally characterized by greater than 10% CSI release [4-1]. As can be seen from the Clinton-specific MAAP calculation, the CSI release fraction is several orders of magnitude lower than the threshold for LERF. This further supports not classifying leakage combination CA'1 as 3b.

TABLE 4.6-1

CPS DWBT AND ILRT LEAKAGE COMBINATION ACCIDENT CLASSES (FROM LAR SUBMITTAL - ATTACHMENT 4)

LEAKAGE COMBINATIONS	DW BYPASS LEAKAGE	CONTAINMENT LEAKAGE	EPRI CLASSIFICATION ASSIGNMENT
AA'	1 DWL_b	1 L_a	1
AB'	1 DWL_b	10 L_a	3a
AC'	1 DWL_b	100 L_a	3b
BA'1	10 DWL_b	1 L_a	1
BB'1	10 DWL_b	10 L_a	3a
BC'1	10 DWL_b	100 L_a	3b
CA'1	**100 DWL_b**	**1 L_a**	**3a**
CB'1	100 DWL_b	10 L_a	3b
CC'1	100 DWL_b	100 L_a	3b

RAI #4 References

[4-1] CPS-PSA-015 CPS Detailed Level 2 Evaluation Notebook, Rev. 2, dated March, 2014.

RAI 5

Question

Similar to the previous one-time ILRT/DWBT frequency extension request, summarize the results of the sensitivity analysis including the probability of drywell failures assigned to small (Class 3a) or large (Class 3b) drywell bypass (DWB) leakage, using the current "as-found" DWBT leakage data for all Mark III containments. Provide the changes (i.e., delta) in large early release frequency (LERF), population dose, conditional containment failure probability, and impact on baseline LERF. The NRC staff notes that the sensitivity documented in Section 6.3 of Attachment 4 to the submittal which increases the probability values of the small (Class 3a) and large (Class 3b) DWB leakage by a factor of 10, does not appear to capture the corresponding probabilities determined based on the historical DWB leakage data using the Chi-square upper bound value.

Response to RAI 5

As discussed in Section 4.6.1 of Attachment 4 of the LAR, CPS is required to vent the drywell approximately once per day to relieve drywell pressure buildup. The need to vent demonstrates a level of drywell integrity that is better than the assumptions used in the existing risk analysis, such that Exelon judges that the existing baseline risk analysis provided in the extension request is conservative. The Clinton drywell risk leakage history presented in Table 4.6-2 of Attachment 4 of the LAR (reproduced in this RAI response as Table 1) is noted to be considerably better than the average suggested by generic Mark III data. Based on the last seven drywell test results, the average for the Clinton drywell is 58 SCFM. The industry average based on test data from non-Clinton Mark III containments is a leakage rate of 901 SCFM. Thus the average Clinton leakage is ~1/15th of the non-Clinton Mark III drywells and indicates that using industry data in a Chi-square upper bound methodology is not representative of Clinton.

Additionally, Section 6.3 of Attachment 4 of the LAR documents a sensitivity case where the assumed DWBT leakage probability is increased an order of magnitude to demonstrate that the conclusions of the ILRT/DWBT risk assessment would not be impacted if the DWBT leakage probability were increased by a factor of ten. In response to this RAI a second bounding sensitivity case is performed using the Chi-square distribution of historical DWB leakage data presented in Table 4.6-2. This new bounding sensitivity case demonstrates that the conclusions of the ILRT/DWBT risk assessment are not impacted.

Chi-Square Upper Bound Sensitivity

This sensitivity case is performed consistent with that previously provided to the NRC for the one time ILRT extension. For consistency with the EPRI methodology, this sensitivity includes impacts associated with liner corrosion.

Historical DWBT Data

Table 1 presents the historical test data for Mark III containment BWRs as presented in Table 4.6-2 of Attachment 4 of the LAR.

The data is interpreted as follows[1] for the Clinton DWBT evaluation:

LEAKAGE RATE	FAILURE CLASSIFICATION
0 – 300 scfm	no "failure"
300 scfm – 3000 scfm	small failure (Category 3a)
>3000 scfm	large failure (Category 3b)

Insights from the Table 5-1 data include the following:

1. No sites measured leakage that would be categorized as large (i.e., greater than 3000 SCFM). Even though there were no observed instances of leakage in this category, the relatively small population of drywell test data will produce a calculated upper bound rate of occurrence for the large category that is larger than was used in the original analysis. The fact that there were no drywell test results in this large category is important because there would have to be large drywell leakage for there to be a change in Large Early Release Frequency. Note that the Clinton risk assessment conservatively categorizes a DW bypass leakage of 10 DWL_b (300 to 3,000 SCFM) combined with a wetwell leakage of 100 L_a as a LERF (EPRI Category 3b) as shown in Table 4.6-1 of the LAR risk assessment.
2. CPS had no tests were leakage exceeded 300 SCFM (i.e., small leakage category). Even though there were no observed instances of leakage in this category, the relatively small population of drywell test data will produce a calculated upper bound rate of occurrence for the small category that is larger than was used in the original analysis.
3. There are 25 completed drywell bypass test results for use.

(1) CPS Threshold (L_b) = 300 scfm
Small Leak Upper Bound ~ 10 L_b = 3,000 scfm
Large Leak characterized as 10 L_b to 100 L_b = (30,000 scfm = 100 L_b)

TABLE 5-1

MARK III DRYWELL BYPASS TEST RESULTS (REPRODUCED FROM TABLE 4.6-2 IN LAR SUBMITTAL RISK ASSESSMENT)

SITE	TEST DATE	LEAKAGE RATE (SCFM)	ACTUAL LEAKAGE / 300 SCFM	>300 SCFM	>3000 SCFM
Clinton	Jan-86	273	0.91	No	No
	Nov-86	20.8	0.07	No	No
	Apr-89	18.8	0.06	No	No
	Mar-91	21.9	0.07	No	No
	May-92	18	0.06	No	No
	Nov-93	30.2	0.1	No	No
	Feb-08	20.18	0.07	No	No
Grand Gulf	Nov-85	2315	7.72	Yes	No
	Nov-86	1568	5.23	Yes	No
	Dec-87	1500	5	Yes	No
	Apr-89	1631	5.44	Yes	No
	Nov-90	1591	5.3	Yes	No
	May-92	618	2.06	Yes	No
	Nov-93	869	2.9	Yes	No
Perry	Aug-87	124	0.41	No	No
	Jul-89	123	0.41	No	No
	Dec-90	797	2.66	Yes	No
	May-92	253	0.84	No	No
	Jun-94	2450	8.17	Yes	No
	Jul-94	111	0.37	No	No
River Bend	Dec-87	602	2.01	Yes	No
	May-89	141	0.47	No	No
	Nov-90	345	1.15	Yes	No
	Aug-92	754	2.51	Yes	No
	Jun-94	421	1.4	Yes	No
			Total "Yes"	**13**	**0**
			Total "No"	**12**	**25**
			Total # Trials	**25**	**25**

Chi-Square Upper Bound Calculation

The sensitivity case for the DWBT interval extension utilizes the 95% confidence Chi-square upper bound based on only Mark III data to characterize the drywell leakage probability. Because only a limited set of data is available for Mark III plants, the statistical analysis of the small sample size could be misleading.

The Chi-square upper bound (95% confidence) estimate is calculated as follows:

$$UB = \frac{X^2\,(0.05, 2f+2)}{2N}$$

Where,

f = # failures

N = # tests

- Large Leak (f = 0)

$$UB = \frac{X^2\,(0.05, 2(0)+2)}{2(25)} = \frac{X^2\,(0.05, 2)}{50} = \frac{5.991}{50} = 0.120$$

- Small Leak (f = 13)

$$UB = \frac{X^2\,(0.05, 2(13)+2)}{2(25)} = \frac{X^2\,(0.05, 28)}{50} = \frac{41.34}{50} = 0.827$$

The small and large leakage categories total 0.947. Note that for this upper bound sensitivity, a leakage probability of almost 1 is calculated. The probability for "no leak" is calculated as:

- No Leak = 1.0 – Large Leak Probability – Small Leak Probability

$$P = 1.0 - 0.120 - 0.827 = 0.053$$

The no leakage category will be assigned a value of 0.053.

The result of the 95% upper bound Chi-square evaluation of the Mark III data is shown in Table 5-2. As can be seen, the drywell bypass probability using the Chi-square 95% upper bound results in an increase in the large bypass probability of approximately a factor of 44 (i.e., 0.120 / 2.7E-3).

Another feature of the results in Table 5-2 merits explanation. Using the Chi-squared upper bound values causes the drywell leakage probabilities to have reduced variance as a function of test frequency. This occurs because, as noted above, use of the Chi-squared results causes the "no leakage" category to have a small (~5%) probability. The probability of being in the "no leakage" category cannot, then, decrease significantly with decreasing test frequency. As a result, the total leakage probability cannot increase substantially with decreasing test frequency. Furthermore, the ratio of large leak probability to small leakage probability is nearly constant. This is because the development of additional large leakage mechanisms would likely be identified by changes in the drywell venting frequency. So, the probabilities of EPRI Categories 3a and 3b for the drywell in this sensitivity study are relatively insensitive to test frequency. These results come from use of the bounding Chi-squared value. However, the combined impact of drywell failure probabilities and containment failure probabilities does still vary with test frequency. As described in the results, below, use of even this highly conservative drywell leakage probability yields containment performance measures that are within acceptable bounds.

TABLE 5-2

DRYWELL BYPASS PROBABILITY BY EPRI CATEGORY

		EPRI METHODOLOGY	CHI-SQUARE UPPER BOUND SENSITIVITY CASE		
	EPRI CATEGORY	3/10 YR	3/10 YR	1/10 YR	1/15 YR
1	No Leak (L_b)	~1	~0.053[1]	~0[1]	~0[1]
3a	Small Leak (10 L_b)	2.70E-2	0.827	0.88[1]	0.88[1]
3b	Large Leak (100 L_b)	2.70E-3	0.120	0.120	0.120

Note to Table 5-2:

(1) The upper bound statistics used produce a conservative result that the drywell leaks with a close to 1 probability. The small leakage probabilities are increased to 0.88 for the 1/10 yr. and 1/15 yr. cases because the total leakage probability cannot exceed 1. The large leakage probabilities for 1/10 yr. and 1/15 yr. are assumed to remain constant, because the development of a new large leakage mechanism would most likely be detected by changes in the CPS drywell daily "burping" frequency.

Tables 5-4 and 5-5 reproduce Table 4.6-4 of Attachment 4 of the LAR incorporating the new drywell bypass probabilities for the 3/10 year frequency (Table 5-4) and the 1/10 year and 1/15 year frequencies (Table 5-5)

Risk Results

The calculations for the combined ILRT and DWBT interval extension are reperformed with the ILRT interval extension characterized as in the main report with the DWBT characterized using the Chi-square upper bound based drywell bypass probabilities.
The results of the sensitivity case analysis are the following when the ILRT/DWBT frequency is reduced from 3/10 year frequency to 1/15 year:

ΔLERF = 1.43E-08/yr

ΔDose Rate = 1.68E-02 Person-Rem/yr

ΔCCDP = 0.64%

These results can be compared with the results of the ILRT/DWBT evaluation using the EPRI methodology. Table 5-3 shows this comparison. The Δ LERF increases by 46%.

TABLE 5-3

COMPARISON OF SENSITIVITY CASE AND EPRI METHOD: CHANGES IN RISK METRICS ASSOCIATED WITH REDUCING THE FREQUENCY OF ILRT/DWBT FROM 3/10 YEARS TO 1/15 YEARS

	Δ DOSE RISK (PERSON-REM/YR)	Δ CCFP	Δ LERF (/YR)
Original EPRI ILRT/DWBT Method	3.80E-03	0.44%	9.81E-9
Chi-Square Upper Bound Sensitivity Case	1.68E-02	0.64%	1.43E-8

As shown in Table 5-3, increases to the population dose and to the CCFP values compared to the base risk assessment, are still within the acceptance criteria of less than 1.0 person-rem/yr and less than 1.5% change in CCFP. The increase in internal events LERF resulting from a change in the Type A ILRT interval and the DWBT interval for the Chi-square sensitivity case with corrosion included is 1.43E-08/yr which falls within the "very small" change region of the acceptance guidelines in Reg. Guide 1.174.

TABLE 5-4

(UPDATED TABLE 4.6-4 FROM LAR RISK ASSESSMENT)
SUMMARY OF THE CONDITIONAL PROBABILITY OF OCCURRENCE
FOR THE POSTULATED LEAKAGE CASES
(DRYWELL BYPASS LEAKAGE PROBABILITY SET BY 95% CHI-SQUARE ESTIMATION)
(CASE NO. 1: ILRT AND DWBT FREQUENCIES AT 3/10 YEARS)

1	2	3	4	5	6	7
LEAKAGE COMBINATIONS	DW BYPASS LEAKAGE	WW LEAKAGE	PROBABILITY OF CASE			EPRI CLASS
			DW	WW	COMBINED	
AA'	1 DWL_b	1 L_a	0.053	0.99	0.0525	1
AB'	1 DWL_b	10 L_a	0.053	0.0092	4.88E-04	3a
AC'	1 DWL_b	100 L_a	0.053	0.0023	1.22E-04	3b
BA'1	10 DWL_b	1 L_a	0.827	0.99	0.819	1
BB'1	10 DWL_b	10 L_a	0.827	0.0092	7.61E-03	3a
BC'1	10 DWL_b	100 L_a	0.827	0.0023	1.90E-03	3b
CA'1	100 DWL_b	1 L_a	0.120	0.99	0.119	3a
CB'1	100 DWL_b	10 L_a	0.120	0.0092	1.10E-03	3b
CC'1	100 DWL_b	100 L_a	0.120	0.0023	2.76E-04	3b
Combined Probabilities (Sum)						
					0.871	1
					0.127	3a
					0.0034	3b

TABLE 5-5

SUMMARY OF THE CONDITIONAL PROBABILITY OF OCCURRENCE FOR THE POSTULATED LEAKAGE CASES (DRYWELL BYPASS LEAKAGE PROBABILITY SET BY 95% CHI-SQUARE ESTIMATION) (CASES NO. 2 AND 3: ILRT AND DWBT FREQUENCIES AT 1/10 YEARS AND 1/15 YEARS)

1	2	3	4	5	6	7
LEAKAGE COMBINATIONS	DW BYPASS LEAKAGE	WW LEAKAGE	PROBABILITY OF CASE			EPRI CLASS
			DW	WW	COMBINED	
AA'	1 DWL_b	1 L_a	0.00	0.99	0.00	1
AB'	1 DWL_b	10 L_a	0.00	0.0092	0.00	3a
AC'	1 DWL_b	100 L_a	0.00	0.0023	0.00	3b
BA'1	10 DWL_b	1 L_a	0.88	0.99	0.871	1
BB'1	10 DWL_b	10 L_a	0.88	0.0092	8.10E-03	3a
BC'1	10 DWL_b	100 L_a	0.88	0.0023	2.02E-03	3b
CA'1	100 DWL_b	1 L_a	0.12	0.99	0.119	3a
CB'1	100 DWL_b	10 L_a	0.12	0.0092	1.10E-03	3b
CC'1	100 DWL_b	100 L_a	0.12	0.0023	2.76E-04	3b
Combined Probabilities (Sum)						
					0.871	1
					0.127	3a
					0.0034	3b

TABLE 5-6

(UPDATED TABLE 6.3-2 OF LAR RISK ASSESSMENT) CPS ILRT/DWBT CASES: 3 IN 10 (BASE CASE), 1 IN 10, AND 1 IN 15 YR INTERVALS (DWBT 10LB, 100LB LEAK PROB. INCREASED FOR CHI-SQ UB)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	2.71E+03	7.94E-07	2.16E-03	4.78E-07	1.30E-03	2.15E-07	6.81E-04
2	5.48E+05	2.68E-07	1.47E-01	2.68E-07	1.47E-01	2.68E-07	1.47E-01
3a	2.71E+04	1.32E-07	3.59E-03	4.41E-07	1.20E-02	6.61E-07	1.80E-03
3b	2.71E+05	3.56E-09	9.67E-04	1.19E-08	3.23E-03	1.79E-0-8	4.86E-03
7 LERF	5.48E+05	1.14E-07	6.27E-02	1.14E-07	6.27E-02	1.14E-07	6.27E-02
7 non-LERF	3.37E+05	9.16E-07	3.09E-01	9.16E-07	3.09E-01	9.16E-07	3.09E-01
8	5.48E+05	1.55E-09	8.49E-04	1.55E-09	8.49E-04	1.55E-09	8.49E-04
Total		2.23E-06	0.526	2.23E-06	0.536	2.23E-06	0.543
ILRT Dose Rate from 3a and 3b		4.56E-03		1.52E-02		2.28E-02	
Delta Total Dose Rate[1]	From 3 yr	---		9.77E-03		1.68E-02	
	From 10 yr	---		---		7.01E-03	
3b Frequency (LERF)		3.56E-09		1.19E-08		1.79E-08	
Delta 3b LERF	From 3 yr	---		8.32E-09		1.43E-08	
	From 10 yr	---		---		6.00E-09	
CCFP %		58.46%		58.83%		59.10%	
Delta CCFP %	From 3 yr	---		0.37%		0.64%	
	From 10 yr	---		---		0.27%	

Note to Table 5-6:

(1) The overall difference in total dose rate is less than the difference of only the 3a, and 3b categories between two testing intervals. This is because the overall total dose rate includes contributions from other categories that do not change as a function of time, e.g., the EPRI Class 2 and 8 categories, and also due to the fact that the Class 1 person-rem/yr decreases when extending the IRLT/DWBT frequency.

Total LERF is calculated below.

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT/DWBT intervals are shown in Table 5-6 as 3.56E-09/yr, 1.19E-08/yr and 1.79E-08/yr, respectively. Using the other hazard group multiplier of 10.31 for CPS, the change in the LERF risk measure due to extending the ILRT/DWBT from 3-per-10 years to 1-per-15 years, including both internal events and other measurable hazard groups hazards risk, is estimated as shown in Table 5-7.

TABLE 5-7
(UPDATED TABLE 5.7-2 OF LAR RISK ASSESSMENT)
CPS 3B (LERF) AS A FUNCTION OF ILRT/DWBT FREQUENCY
FOR INTERNAL AND EXTERNAL EVENTS
(INCLUDING AGE ADJUSTED STEEL CORROSION LIKELIHOOD
AND CHI-SQUARE UB)

	3B FREQUENCY (3-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-15 YEAR ILRT/DWBT)	LERF INCREASE(1)
Internal Events Contribution	3.56E-09	1.19E-08	1.79E-08	1.43E-08
Other Hazard Group Contribution (Internal Events CDF x 10.31)	3.67E-08	1.23E-07	1.85E-07	1.47E-07
Combined	4.03E-08	1.35E-07	2.02E-07	1.62E-07

Note to Table 5-7 (5.7-2):

(1) Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

Thus, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution using the Chi-square upper bound is estimated as 1.62E-07/yr, which includes the age adjusted steel corrosion likelihood.

TABLE 5-8

(UPDATED TABLE 5.7-4 OF LAR RISK ASSESSMENT) IMPACT OF 15-YR ILRT EXTENSION ON LERF (3B) FOR CPS USING CHI-SQUARE UB

Internal Events LERF	1.16E-07/yr
Internal Fire LERF	9.21E-07/yr
Other Hazard Group LERF (Internal Events LERF x 7.62)	8.84E-07/yr
Internal Events LERF due to ILRT (Class 3b) at 15 years (1)	**1.79E-08/yr**
Other Hazard group LERF due to ILRT at 15 years (1)	**1.85E-07/yr**
Total	**2.12E-06/yr**

Note to Table 5-8:

(1) Including age adjusted steel corrosion likelihood.

As can be seen, the estimated LERF for CPS using the Fire LERF and CDF based multiplier approach for seismic and the Chi-square upper bound is 2.12E-06/yr, which is less than the RG 1.174 required value of 1E-5/yr. This compares to the baseline LERF total of 2.06E-06/yr as shown in Table 5.7-4 of the LAR risk assessment.

RAI 6

Question

Section 5.7 of Attachment 4 to the submittal, states that other external hazards, including high winds and tornadoes, external floods, transportation accidents, and nearby facility accidents were not considered because of their negligible contribution to overall plant risk. This conclusion was reached based on the CPS Individual Plant Examination for External Events (IPEEE) analysis. Since the IPEEE studies have not been recently updated, discuss the applicability of the IPEEE conclusions with regards to each of the above-mentioned hazards to the current plant configuration and operating experience, and taking into account updated risk studies and insights.

Response to RAI 6

High Winds and Tornadoes (including Tornado Missiles)

The Clinton IPEEE reviewed the as built plant against the requirements of the SRP, with respect to high winds, tornadoes and tornado missiles. The IPEEE determined that the plant met the SRP with respect to wind and tornado loading. The design basis tornado (DBT) for Clinton had winds of 360 mph, based on RG 1.76. Clinton also met the requirements of the SRP, Section 3.5.1.4 with respect to tornado missiles.

A review of the current plant condition and any updated wind or tornado hazards was performed.

- High Winds and Tornado Winds - Since the IPEEE, RG 1.76 was updated to lower the required Design Basis Tornado (DBT) wind speeds (to 230 mph for the region in which Clinton is located). Since Clinton was designed to a higher wind speed, the risk associated with the pressure and differential pressure effects of high winds and tornadoes remains negligible, and the conclusions of the IPEEE are not changed.
- Tornado Missiles – The Clinton tornado missile analysis (using TORMIS) was updated in 2007. The results of the analysis showed that the total damage frequency of safety related SSCs exposed to tornado missiles is less than 10^{-6}/yr for current conditions. Therefore the risk from tornado missiles is negligible and the conclusions of the IPEEE are not changed.

The review of the current plant condition and local/regional wind and tornado hazards indicate that high winds and tornadoes at Clinton can still be screened, based on low risk to the plant.

External Floods

Clinton Power Station (CPS) reevaluated the external flood hazard in accordance with NTTF Rec. 2.1 and submitted the results to NRC on March 12, 2014 as a Flood Hazard Reevaluation Report (FHRR) [1]. The FHRR utilized current day methodology for evaluating the flooding hazard for new reactor sites. Criteria used for this evaluation can be found in NUREG/CR-7046 [2] and employs a Hierarchical Hazard Analysis (HHA) approach where conservative assumptions inputs and methods (AIMs) are used to determine if any impact to the plant will be realized from the flood hazard. Refinements and the introduction of more realistic AIMs are used when a more conservative approach proves challenging.

CPS chose to utilize the most conservative AIMs in their FHRR and determined that the current design basis (CDB) is not exceeded by the new reactor standards for determining the flood hazard at the station. The following flooding mechanisms for CPS were evaluated in the FHRR and determined to be bound by the CDB:

1. Local Intense Precipitation (Bounded)
2. Flooding in Streams and Rivers (Bounded)
3. Probable Maximum Flood on Cooling Lake (Bounded)
4. Upstream Dam Failure (N/A)
5. Storm Surge & Seiche (N/A)
6. Tsunami (N/A)
7. Ice Induced Flooding (Bounded)
8. Channel Migration or Diversion (N/A)
9. Low Water (N/A)
10. Combined Effects (N/A)

Therefore, it is reasonable to conclude that the insights and analyses provided in the IPEEE have not changed nor need to be updated. The conservative AIMs used for the FHRR demonstrates that maximum flood levels from the applicable mechanisms above would not exceed the CDB and it has been postulated that these maximum flooding events are extremely unlikely. There is no evidence to suggest a change to the overall conclusion that the risk from external flooding is low.

References

[1] Exelon Generation Company, LLC Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 2, Flood Hazard Reevaluation Report (FHRR), March 12, 2014, NRC Docket No. STN 50-461.

[2] U.S. Nuclear Regulatory Commission. NUREG/CR-7046, PNNL-20091, Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America. November 2011.

Transportation and Nearby Facility Accidents

The Clinton IPEEE report reviewed the environment around Clinton Power Station to try to identify Nearby Facility Accidents that pose a significant risk to CPS. The salient points from that study are:

- Clinton Power Station is in an area of low industrial activity with only 3 facilities within 5 miles of the plant that stored hazardous materials (one of them being the plant itself).
- The only substance of sufficient quantity, volatility and toxicity to pose a toxic hazard to Clinton Power Station was anhydrous ammonia used for fertilizer stored at an agricultural supply facility in DeWitt Illinois about 2 miles from CPS. The risk of a release incapacitating Main Control Room operators from this facility was demonstrated to be acceptably low.
- One of the facilities in DeWitt had propane for resale, but given the distance from the plant, had substantially less than the quantity of propane that would pose an explosive hazard to the plant.
- Chemicals stored onsite at the Clinton Plant were found not to pose a threat to the safety of the plant.

Based upon a 2013 survey of Hazardous Chemicals for Main Control Room Habitability the region within 5 miles of the site has not changed appreciably with regard to the quantities of hazardous materials stored in offsite facilities, with limited quantities of propane, anhydrous ammonia and nitrogen fertilizer (the later in aqueous form making it much less volatile than anhydrous ammonia). The facility storing anhydrous ammonia is the same one identified in the IPEEE study. The impact of an anhydrous ammonia release from this facility on Main Control Room habitability has been evaluated and was found to be acceptable. Chemicals stored onsite at Clinton Power Station were also reviewed and none of them were identified as posing a threat to plant safety, generally because of low toxicity or volatility.

With regard to highway transportation accidents by highway the IPEEE made the following points:

- While there are 3 two lane state highways within 5 miles of the site, there are no interstate highways near the plant. Also the pattern of interstate and US highways connecting major cities in the area (i.e. Champaign-Urbana, Bloomington-Normal, Decatur and Springfield) make use of these state highways illogical routes for transportation of materials, that do not originate or are destined, for locations within the perimeter of these cities.
- Therefore highway traffic near the site involving hazardous materials was identified by surveying nearby facilities (in a somewhat larger area in DeWitt County) regarding the types of hazardous materials they stored and if necessary the highway routes they used.
- A number of methods we used to pare down the list of chemicals considered for transportation hazards including quantity stored, toxicity and volatility.
- Anhydrous ammonia and chlorine (for water treatment) appeared in the area and were considered because of their toxicity and volatility. The later was determined not to be shipped near Clinton Power Station.
- Anhydrous ammonia shipped to the agricultural facility in DeWitt was evaluated for its risk impact to the Main Control Room by giving consideration to the number of shipments, shipping route and quantity of ammonia shipped in a probabilistic study. The risk from transported anhydrous ammonia was summed with that from the fixed storage facility and was found to be acceptable.

The 2013 survey of Hazardous Chemicals for Main Control Room Habitability also looked at the hazards from highway transportation accidents using a survey approach for local facilities storing hazardous material. Propane and anhydrous ammonia were found to be shipped past the site from this survey. The amount of propane shipped per truck was found to be below the quantity that would be capable of damaging Clinton Power Station at the distances involved. The shipments of anhydrous ammonia to the facility in DeWitt were found to pass on highway 54, which at one point passes close to the site. But the number of such shipments was less than 10 per year which eliminated it from further consideration.

While the IPEEE did not reevaluate risk of railroad transportation accidents, because of a Clinton Power Station licensing commitment to periodically survey the rail traffic on the Gilman Line passing near Clinton Power Station, the more recent 2013 survey of Hazardous Chemicals for Main Control Room Habitability did look at hazardous materials shipped by rail near the site. While there were several hazardous chemicals shipped, none exceeded 30 shipments per year which was a screening criteria allowing them to be screened from further consideration.

The more recent work considering Transportation and Nearby Facility Accidents has not provided any reason to conclude that the risk of these accidents has increased to the point that these accidents would significantly impact the ILRT extension application.

RAI 7

Question

The licensee has discussed its Fire PRA in Section A.3.1 in Appendix A of Attachment 4 to the submittal. Address the following regarding the fire PRA:

a. Address, preferably quantitatively such as through sensitivity analyses, whether the estimated fire CDF (6.0E-6/yr) and LERF (9.21E-7/yr) are bounding with respect to the current state-of-the-art for fire PRA considering all approved guidance since NUREG/CR-6850 was first issued.

b. Address whether any "unapproved/unreviewed analysis methods" were employed in the current application of the Fire PRA.

Response to RAI 7a

The 2014 CPS Fire PRA CDF of 6.0E-6/yr and LERF of 9.21E-7/yr are judged to be reasonable estimates of the Fire CDF risk for the CPS for the current state of knowledge. Section A.3.2 in Appendix A of Attachment 4 to the submittal identifies limitations of the fire model and notes that model conservatisms, such as assuming fire induced failure of balance of plant components in many plant locations, are judged to compensate for model non-conservatisms such as lack of multi-compartment fire scenarios.

FPRA Guidance

With regards to Fire PRA methodology guidance approved subsequent to the issuance of NUREG/CR-6850 in 2005, the following is noted:

- Guidance included in Supplement 1 to NUREG/CR-6850 (2009) was incorporated in the 2014 Fire PRA as follows:
  - FAQ 06-16 (Ignition Source Counting Guidance for Electrical Cabinets) – EP counting is judged to be consistent with the clarifications provided in the guidance.
  - FAQ 06-17 (Ignition Source Counting Guidance for HEAFs) – FIF counting differentiates low and medium voltage HEAF sources consistent with the guidance.
  - FAQ 06-18 (Ignition Source Counting Guidance for Main Control Boards) – Main Control Room ignition sources were binned as directed in the guidance.

- FAQ 07-31 (Miscellaneous Binning Issues) – Ignition source counting is judged to conform to the ignition source size criteria established in the guidance.
- FAQ 07-35 (Bus Duct Counting Guidance for HEAFs) – Simplified bus duct scenarios were added for the 2014 FPRA update to represent a fire on the RAT/ERAT bus ducts that connect to the safety/non-safety switchgears. Bus duct counting was performed at a room level (i.e., one count per room containing bus duct of a given type) rather than using a linear or segment approach discussed in the guidance. Also, the impact of the fire scenarios was limited to the bus duct itself, which would fail the RAT/ERAT. These simplified bus duct scenarios are estimated to be somewhat non-conservative since damage below the bus duct is not modeled. These scenarios are planned to be refined in the next FPRA update using the FAQ 07-35 guidance as supported by walkdowns.
- FAQ 08-42 (Fire Propagation from EPs) – All electrical panels (EP) are conservatively assumed to be unsealed for the FPRA and therefore capable of propagating outside the EP. Therefore the potential credit for robustly secured EPs is not taken.
- FAQ 08-44 (Main FW Pump Oil Fires) – FW Pump oil fires are developed based on the categories presented in the guidance.
- FAQ 08-48 (Fire Ignition Frequency) – The updated generic bin frequencies developed in the guidance are used in the 2014 FPRA.
- FAQ 08-49 (Cable Tray Fire Propagation) – The cables at CPS are IEEE-383 qualified and are assumed to exhibit thermoset material properties. Cables are assumed damaged by fire at the minimum failure temperature. Conservative zones of influence (ZOIs) are used based on 98% HRRs to estimate fire damage based on Generic Fire Modeling Treatments. The conservative ZOIs are assumed to compensate for more specific analysis of cable fire propagation and other secondary combustibles.
- FAQ 08-43 (Fire Location in EPs) – For the application of ZOIs, the fire is assumed to be at the top of the EP. This may be somewhat conservative for some panels since the guidance allows locating the fire at 1' below the top of the panel.
- FAQ 08-46 (Incipient Fire Detection) – The FPRA does not credit incipient fire detection so this FAQ does not apply.
- FAQ 08-50 (Manual Nonsuppression Probability) – Manual suppression is only explicitly credited for Main Control Room (MCR) fires. The probability for manual suppression of MCR fires uses the mean suppression rate (lambda) value of 0.33 which is the same between NUREG/CR-6850 Appendix P and this FAQ guidance.

- FAQ 08-47 (Spurious Operation Probability) – For the 2014 FPRA, spurious operation probabilities are based on the values presented in NUREG/CR-7150 Volume 2. Aggregate values are conservatively used. For some less risk significant components, a spurious operation value of 1.0 may be conservatively assumed. Limited spurious duration is modeled for select components (e.g., MSIVs, containment isolation valves) following the guidance of NUREG/CR-7150.
- FAQ 08-52 (Transient Fire Growth Rate) – Consistent with the guidance, transient fires in the MCR use the MCR fire non-suppression curve. With regards to specific transient fire growth rates, the Generic Fire Modeling Treatments used to develop the ZOIs assume a fully developed fire for a given HRR level such that a specific growth rate is not used. For the MCR abandonment time evaluation, a composite transient fire growth rate of approximately 4 minutes was used.

- Additional guidance issued subsequent to the publication of Supplement 1 to NUREG/CR-6850 was also incorporated:
  - FAQ 12-64 (Fire Ignition Frequency Clarifications) – Transient influence factors have been revised to establish a normative value of "Medium", as prescribed by the guidance.
  - FAQ 13-04 (Sensitive Electronics) – As discussed in the guidance, internally mounted sensitive electronics are addressed using the heat flux damage threshold for thermoset cables given that the cabinet enclosure provides radiant heat protection. Panels were not opened to determine the exact mounting location of any sensitive electronics. For fires within any portion of an electrical panel, all sections of the electrical panel are conservatively failed. Externally mounted sensitive electronics were not found during walkdowns.
  - FAQ 13-05 (Self-Ignited Cable Fires) – Since CPS employs IEEE-383 qualified cables, self-ignited cable fires were not postulated.
  - FAQ 13-06 (Junction Box Scenarios) - Junction box scenarios were not postulated in the 2014 FPRA. This lack of inclusion is judged to be minimally non-conservative. As the guidance notes, in most cases such fires do not generate enough heat to become self-sustaining and will self-extinguish prior to spreading outside the junction box. Since CPS employs IEEE-383 cables, the potential for fire spread is minimized.
  - NUREG-1921 (EPRI/NRC-RES Fire Human Reliability Analysis Guidelines) was issued in 2012 to provide more explicit guidance for FPRA HRA. The 2014 CPS Fire HRA employs guidance from NUREG-1921 Appendix C in developing the detailed HEPs.

Sensitivity Analysis

Developing individual sensitivity analyses for the different potential model limitations is not a simple endeavor, and would not necessarily reflect how the differing model limitations might interact. Given the large margin of the ILRT risk assessment results to the acceptance criteria, a single sensitivity case is presented to demonstrate that the Fire CDF and Fire LERF would need to increase almost an order of magnitude in order to exceed the risk assessment acceptance criteria.

A sensitivity is performed assuming Fire CDF and LERF estimates are increased by a factor of nine (9). The License Amendment Request, Attachment 4, Section 5.7 is updated, to show the impact of increasing Fire CDF and Fire LERF by a factor of nine.

ESTIMATED CDF (x9)	ESTIMATED LERF (x9)
5.4E-5/yr	8.29E-6/yr

Sensitivity Summary Result

As shown in Table 5.7-3 (x9 Fire Sensitivity) *CPS 3b (LERF) as a Function of ILRT/DWBT Frequency for Internal and External Events*:

- The LERF increase associated with permanently extending the ILRT surveillance interval to 15 years is 3.22E-7/yr and below the acceptance criteria of <1.0E-6/yr.
- The increase in Person-Rem/yr of 1.25E-01/yr (0.73%) is below the acceptance criteria of <1.0 person-rem/yr or <1.0%.

The Fire x9 sensitivity shows a nine times increase in estimated Fire CDF and LERF does not exceed acceptance criteria and would not alter the conclusions of the ILRT LAR submittal. This demonstrates significant margin that is available for the Fire PRA results.

Sensitivity Based Update to LAR Text

The following presents a revised portion of LAR Attachment 4, Section 5.7, with the text updated to reflect the sensitivity analysis for a nine fold increase in Fire CDF and LERF.

Other Hazard Group Contributor Summary (Updated for x9 Fire CDF/LERF Sensitivity)

The method chosen to account for external events contributions is similar to that used in the other ILRT interval extension analyses [28, 39] in which a multiplier is applied to the internal events results. The contributions of the external events from various CPS analysis are summarized in Table 5.7-1 (x9 Fire Sensitivity).

TABLE 5.7-1 (X9 FIRE CDF SENSITIVITY)

OTHER HAZARD GROUP CONTRIBUTOR SUMMARY

OTHER HAZARD INITIATOR GROUP	CDF (1/YR)
Seismic [9]	1.7E-05
Internal Fire [8] **(x9 Fire Sensitivity)**	**5.4E-05**
High Winds/Tornadoes	Screened
External Floods	Screened
Transportation and Nearby Facility Accidents	Screened
Total (for initiators with CDF available)	**7.1E-05**
Internal Events CDF	2.23E-06
External Events Multiplier	**31.8**(1)

Note to Table 5.7-1:

(1) The multiple for seismic alone is 7.62. **(x9 Fire CDF is a multiple of 24.2)**

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT/DWBT intervals are shown in Table 5.6-1 as 2.43E-09/yr, 8.12E-09/yr, and 1.22E-08/yr, respectively. Using the other hazard group **(x9 Fire Sensitivity)** multiplier of **31.8** for CPS, the change in the LERF risk measure due to extending the ILRT/DWBT from 3-per-10 years to 1-per-15 years, including both internal events and other measurable hazard groups hazards risk, is estimated as shown in Table 5.7-2.

TABLE 5.7-2 (X9 FIRE SENSITIVITY)

CPS 3B (LERF) AS A FUNCTION OF ILRT/DWBT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS (INCLUDING AGE ADJUSTED STEEL CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-10 YEAR ILRT/DWBT)	3B FREQUENCY (1-PER-15 YEAR ILRT/DWBT)	LERF INCREASE(1)
Internal Events Contribution	2.43E-09	8.12E-09	1.22E-08	9.81E-09
Other Hazard Group Contribution (Internal Events CDF x **31.8**)	**7.73E-08**	**2.58E-07**	**3.88E-07**	**3.12E-07**
Combined	**7.97E-08**	**2.66E-07**	**4.00E-07**	**3.22E-07**

Note to Table 5.7-2:

(1) Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

Thus, the total increase in LERF (measured from the baseline 3-per-10 year ILRT interval to the proposed 1-per-15 year frequency) due to the combined internal and external events contribution **for the x9 Fire Sensitivity** is estimated as **3.22E-07/yr**, which includes the age adjusted steel corrosion likelihood.

The other metrics for the ILRT/DWBT interval extension risk assessment can be similarly derived using the multiplier approach. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are shown in Table 5.7-3 **(x9 Fire Sensitivity).** As can be seen, the impacts from including the other hazard group contributors are as follows:

1. Change in LERF = **3.22E-7/yr**, which is above the 1.0E-7/yr upper boundary for the "very small" risk increase as defined in RG 1.174, but in bottom portion of the band for "small" risk increase.
2. Change in population dose rate is 1.25E-1 person-rem/yr (0.73%), which is less than 1.0 person-rem/year or 1% of the total population dose.
3. Change in CCFP is 0.44%, which is less than 1.5%.

Thus, increasing the fire risk by a factor of nine does not change the conclusion of the risk assessment. That is, the acceptance criteria are sufficiently met such that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years has been demonstrated to be small. Note that a bounding analysis for the total LERF contribution follows Table 5.7-3 to demonstrate that the total LERF value for CPS is less than 1.0E-5/yr consistent with the requirements for a "Small Change" in risk of the RG 1.174 acceptance guidelines.

TABLE 5.7-3 (X9 FIRE SENSITIVITY)

COMPARISON TO ACCEPTANCE CRITERIA INCLUDING OTHER HAZARD GROUPS CONTRIBUTION FOR CPS

CONTRIBUTOR	ΔLERF	ΔPERSON-REM/YR(1)	ΔCCFP(2)
CPS Internal Events	9.81E-9/yr	3.80E-03/yr (0.73%)	0.44%
CPS Other Hazard Groups	**3.12E-7/yr(3)**	**1.21E-01/yr(3) (0.73%)**	0.44%
CPS Total	**3.22E-7/yr**	**1.25E-01/yr (0.73%)**	0.44%
Acceptance Criteria	<1.0E-6/yr	<1.0 person-rem/yr or <1.0%	≤1.5%

Notes to Table 5.7-3:

(1) The EPRI Class (1, 2, 7, 8) release Person-Rem/yr are assumed to be the same percentage relative to base risk (0.73%) for internal and external events.

(2) The Probability of DW and WW leakage due to the ILRT/DWBT extension is assumed the same for both Internal and External Events, therefore the percentage change for CCFP remains constant (0.44%).

(3) Value calculated as the Internal Events value x 31.8 external event multiplier.

The **3.22E-07/yr** increase in total LERF **for the x9 fire sensitivity** due to the combined hazard events from extending the CPS ILRT/DWBT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1E-7 to 1E-6 per reactor year ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the "Small Change" range, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-2, the total LERF due to postulated internal event accidents is the sum of the LERF release categories, which is 1.16E-7/yr. For Fire **for the x9 sensitivity**, the total Fire LERF is increased to **8.29E-6/yr**. The base LERF due to seismic is assumed to be in the same proportion as the internal events contribution. The total LERF value for CPS is then shown in Table 5.7-4.

TABLE 5.7-4

IMPACT OF 15-YR ILRT EXTENSION ON LERF (3B) FOR CPS (X9 FIRE SENSITIVITY)

Internal Events LERF	1.16E-07/yr
Internal Fire LERF	**8.29E-06/yr**
Other Hazard Group LERF (Internal Events LERF x 7.62)	8.84E-07/yr
Internal Events LERF due to ILRT (Class 3b) at 15 years (1)	1.22E-08/yr
Other Hazard group LERF due to ILRT at 15 years (1)	**3.88E-07/yr**
Total	**9.69E-06/yr**

Note to Table 5.7-4:

(1) Including age adjusted steel corrosion likelihood.

As can be seen, the estimated total LERF for CPS assuming a Fire LERF that is nine times the 2014 Fire PRA remains below the RG 1.174 threshold value of 1E-5/yr. This quantitative sensitivity demonstrates a significant margin exists with respect to the potential increase in fire risk metrics required to change the conclusions of this risk assessment.

Response to RAI 7b

As discussed in Section A.3.1 in Appendix A to the LAR, the 2014 CPS Fire PRA is an interim implementation of NUREG/CR-6850 and not all tasks were completely performed (e.g., multi-compartment analysis). The response to RAI 7a provides additional detail regarding the 2014 FPRA and guidance subsequent to issuance of NUREG/CR-6850. The FPRA has not received a Peer Review. During the development of the 2014 PRA, accepted methods were applied using industry guidance and consistent with other Exelon peer reviewed FPRAs.

RAI 8

Question

Section 5.3 in Attachment 4 to the submittal, states that Class 7 sequences are impacted by the ILRT/DWBT interval extension. The statement appears to be inconsistent with arguments in Section 4.1.3, which explains that Class 7 sequences are not impacted by the requested extension. As Tables 5.2-2, 5.3-1 and 5.3-2 do not show any impact on Class 7, clarify the treatment of Class 7 sequences.

Response to RAI 8

Section 4 EPRI Class 7 Sequence Treatment

The argument in Section 4.1.3 is targeted to over pressurization challenges that could potentially lead to loss of containment due to pressurization beyond the containments ultimate capability thereby impacting Class 7 sequences. The first paragraph of Section 4.1.3 summarizes that while other Mark III analyses have conservatively assumed that increase drywell bypass leakage would lead to higher containment pressures and containment failures (captured in Class 7), CPS performed MAAP calculations to examine this potential and found negligible potential impacts. The containment overpressure scenario is discussed in detail in Section 4.6.1 DWBT Data Analysis section of Attachment 4 to the submittal. Justification for not increasing CDF begins on page 4-61of the submittal. The justification concludes on page 4-63 of the submittal with the following:

> "Based on the significant margin found in the 2003 LAR MAAP runs and the deterministic arguments noted above, the following conclusion is reached for the request for a permanent 15 interval risk assessment:
>
> There is no change in CDF due to the small increases in drywell bypass leakage associated with the DWBT interval extension."

Section 5 Class 7 Sequence Treatment

The Section 5 statements regarding DWBT intervals impacting Class 7 were unintentionally misleading. As discussed in Section 4, although drywell bypass leakage has the potential to lead to higher containment pressures and containment failures, CPS specific MAAP runs determined this potential to be of negligible importance. As a result, Table 5.2-2, 5.3-1, and 5.3-2 are correct in not showing any impact on Class 7 for changes in ILRT/DWBT frequency.

Conclusions

The EPRI Class 7 sequences are not impacted by a change in the ILRT/DWBT frequency as evaluated specifically for CPS.