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Withhold from Public Disclosure in accordance with 10 CFR 2.390. Upon removal of Enclosure A, this Letter is uncontrolled.

10 CFR 50
10 CFR 51
10 CFR 54

RS-14-266

September 11, 2014

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Withdrawal and Resubmittal of Information associated with NRC Set 10 RAIs, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application

References:

1. Letter from Michael P. Gallagher, Exelon Generation Company (Exelon) to NRC Document Control Desk, dated July 15, 2014, "Resubmittal of Information associated with NRC Set 10 RAIs, related to the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, License Renewal Application"

2. Letter from Lindsay R. Robinson, US NRC to Michael P. Gallagher, Exelon, dated September 3, 2014, "Request for Withholding Information from Public Disclosure (TAC Nos. MF1879, MF1880, MF1881, MF1882)"

Reference 1 describes the history associated with Exelon Generation Company, LLC (Exelon) submitting the License Renewal Application (LRA) for the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (BBS), the NRC staff issuing the Set 10 Requests for Additional Information (RAI), and several additional letters involving these Set 10 RAIs. Reference 1 also provides a Reference listing for these correspondences.

Within Enclosures A and B of Reference 1, Exelon re-submitted proprietary and non-proprietary versions, respectively, of the responses to the ten Set 10 RAIs that were believed to contain proprietary information. However, as described in Reference 2, the NRC staff reviewed this information and concluded that, in some cases, insufficient justification was provided to determine that the information sought to be withheld from public disclosure contains proprietary information.

Based upon this feedback, the Set 10 responses have been re-evaluated. Exelon requests withdrawal of these responses that were provided in Enclosures A and B of Reference 1.

To replace these Set 10 RAI responses, Exelon provides the following:

1. Within Enclosure A – Responses to Requests for Additional Information containing Proprietary Information, based on Westinghouse letter LTR-PAFM-14-31, Rev. 3, Attachment 1, "Byron and Braidwood Units 1 and 2 License Renewal: NRC Request for Additional Information Responses (Proprietary)"

2. Within Enclosure B - Responses to Requests for Additional Information with Proprietary Information redacted, based on Westinghouse letter LTR-PAFM-14-31, Rev. 3, Attachment 2, "Byron and Braidwood Units 1 and 2 License Renewal: NRC Request for Additional Information Responses (Non-Proprietary)"

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Enclosure C of this letter provides the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-14-4026, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

Accordingly, it is requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the item listed above or the supporting Westinghouse Affidavit should reference CAW-14-4026 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry, Pennsylvania 16066.

There are no new or revised regulatory commitments contained in this letter.

If you have any questions, please contact Mr. Al Fulvio, Manager, Exelon License Renewal, at 610-765-5936.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 9-11-2014

Respectfully,

Michael P. Gallagher

Michael P. Gallagher
Vice President - License Renewal Projects
Exelon Generation Company, LLC

Enclosures: A: Responses to Requests for Additional Information (Proprietary)
B: Responses to Requests for Additional Information (Non-Proprietary)
C: Application for Withholding Proprietary Information from Public Disclosure

cc: Regional Administrator – NRC Region III
NRC Project Manager (Safety Review), NRR-DLR
NRC Project Manager (Environmental Review), NRR-DLR
NRC Senior Resident Inspector, Braidwood Station
NRC Senior Resident Inspector, Byron Station
NRC Project Manager, NRR-DORL-Braidwood and Byron Stations
Illinois Emergency Management Agency - Division of Nuclear Safety

Enclosure B

Byron and Braidwood Stations (BBS), Units 1 and 2
License Renewal Application
Responses to Requests for Additional Information

Non-Proprietary Responses

RAI 4.3.1-1
RAI 4.3.1-3
RAI 4.3.4-1
RAI 4.3.4-2
RAI 4.3.4-3
RAI 4.3.4-4
RAI 4.3.4-5
RAI 4.3.4-6
RAI 4.3.4-7
RAI 4.3.7-1

Notes:

1. The responses contained in this Enclosure do not contain proprietary information. Such information has been redacted from these responses as evidenced by the blank space within the brackets shown within the responses.

2. As further explained in the Proprietary Information Notice and Affidavit contained in Enclosure C, the justification for considering certain information proprietary is indicated by means of lower case letters located as a superscript adjacent to the brackets identifying each proprietary item. These lower case letters correspond to Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit.

RAI 4.3.1-1, Transient basis redefinition for 4 new transients (060)

Applicability: Byron Nuclear Station (Byron) and Braidwood Nuclear Station (Braidwood)

Background:

The license renewal application (LRA) states that for Transient 6, "Letdown Flow Shutoff Prompt Return to Service," in LRA Table 4.3.1-2 and 4.3.1-5, the baseline cycles for Byron Unit 2, and the projected 60-year cycles for all four units exceed the current licensing basis (CLB) cycle limit of 200. The LRA states that the transient was redefined as four differential temperature range transients. The LRA further states that number of baseline and 60-year projected cycles for each of the differential temperature range transients were determined and a reanalysis was performed for the bounding location, which confirmed that the cumulative fatigue usage will remain below 1.0.

Issue:

The staff is unclear on the technical basis for redefining the original transient definition to four new transients. The applicant did not provide the new transient definitions, baseline cycle counts, 60-year projected cycle counts, and CLB cycle limits for the four redefined transients.

Request:

1. Provide the four redefined differential temperature range transient definitions and identify the cycle limits, the baseline cycle counts, and projected cycle counts for each new transient.

2. Describe and justify the basis for redefining the original transient definition.

3. Update the applicable LRA tables to include the redefined transients.

4. Confirm that the Fatigue Monitoring Program, when implemented, will monitor the redefined transient cycles and severities and will require action prior to exceeding design limits. If not, justify that the aging effects due to fatigue will be managed during the period of extended operation for the components impacted by these redefined transients.

Exelon Response:

1. Each unit has both a normal and an alternate charging line. The use of these lines is alternated each refueling outage prior to putting the Chemical Volume and Control System in service. The purpose of alternating the use of the lines is to distribute the fatigue effects of system transients between each of the lines. The charging nozzle is the limiting component on each line. Initially, conservative values of both baseline and projected transients were developed for license renewal. Some of the conservative values resulted in exceeding the CLB cycle limit. In addition, since this nozzle is within the reactor coolant pressure boundary and is a NUREG/CR-6260 location, the effects of environmentally assisted fatigue were evaluated for license renewal. To evaluate this condition further, review of the transient history was conducted as described in the response to Request 2 below. The four redefined differential temperature range transient definitions along with the

corresponding period of extended operation (PEO) cycle limits, baseline cycle counts, and 60-year projected cycle counts are provided in the following table. The baseline and projected cycles for the redefined differential temperature transient cases are per charging nozzle (normal or alternate). To provide a conservative fatigue evaluation, instead of using the 60-year projected cycles for each nozzle, which would be 50% of the combined total cycles, the fatigue evaluation of each nozzle used input cycle values that were [ ]a,c,e of the 60-year projected total cycles for both nozzles. It should be noted the values in LRA Tables 4.3.1-2 and 4.3.1-5 for transient 6, “Letdown Flow Shutoff Prompt Return to Service,” represent total cycles, not cycles per nozzle.

Letdown Flow Shutoff Prompt Return to Service Redefined Transient Baseline and 60-Year Cycle Projections per Charging Nozzle

Differential Temperature Transient Cases	Byron Station, Unit 1 (1)		Byron Station, Unit 2 (1)		Braidwood Station, Unit 1 (1)		Braidwood Station, Unit 2 (1)		EAF Evaluation Cycles (PEO CLB Cycle Limits) (2)
	Baseline Cycles	Projected Cycles	Baseline Cycles	Projected Cycles	Baseline Cycles	Projected Cycles	Baseline Cycles	Projected Cycles	
Case A: [ ]a,c,e	20	35	27	44	21	41	17	33	70
Case B: [ ]a,c,e	58	96	76	123	59	115	47	94	180
Case C: [ ]a,c,e	5	6	6	8	5	8	4	6	15
Case D: [ ]a,c,e	10	16	13	20	10	19	8	15	25

Note 1: The baseline and projected cycles presented above for Case A, Case B, Case C, and Case D of the Letdown Flow Shutoff Prompt Return to Service transients are applicable to each charging nozzle (normal and alternate). These values are 50% of the combined total cycles determined from plant monitoring data. This is consistent with the design basis assumption that the normal and alternate charging nozzles are used alternately.

Note 2: EAF Evaluation Cycles, which will become PEO CLB Cycle Limits, are limits for each charging nozzle.

2. Comparison of actual plant-specific transient data against the original design transients demonstrated that operating transients were less severe than the design transients, even though total cycles were greater. Therefore, it was appropriate to redefine and analyze the transient based on smaller enveloping severities that bounded the operating transients.

   Plant computer data of actual plant transients was reviewed for the period of 1999 through 2011 in the operating charging line of each of the units. The parameters that were reviewed included regenerative and letdown heat exchanger outlet temperatures, charging and letdown flows, and reactor coolant loop temperatures. Single thermal excursions were characterized by magnitude and rate of temperature change, among other characteristics. Review of the plant data showed a large number of transient cycles associated with flow isolation design transients with temperature changes (ΔT) that were well below the ΔT associated with the design transients but exceeded the number of cycles assumed in original design analysis. The cycles were counted within various bounding ΔT ranges for each unit. The EAF Evaluation Cycles (PEO CLB cycle limits) for each ΔT range were established to envelop all four units based on the Byron Unit 2 data, which has the maximum projected cycles in each ΔT range. This distribution was then applied to all four units.

The baseline cycle numbers prior to 1999 and from 2011 through 2012 are based on extrapolation of a large amount of plant operating transient data related to the normal and alternate charging lines from 1999 to 2011. Charging line operating procedures have not changed significantly since 1986, confirming the charging lines have been operated consistently from initial plant startup until 2011, and each charging line was alternated each refueling outage. These conclusions were confirmed during plant operator interviews. Therefore the transient data that was collected and reviewed from 1999 to 2011 is representative for the period from initial plant startup to 1999 and from 2011 through 2012.

3. As shown in response to Request 1, the detailed definitions of the redefined transients are considered proprietary. Note 8 in Tables 4.3.1-2 and 4.3.1-5 of the LRA provides a high level description of the redefined transients. Therefore, an update to the LRA tables to include the redefined transients cannot be made and is considered not necessary.

4. The transients and limits documented in the above table will be monitored by Fatigue Monitoring (B.3.1.1) aging management program implementing procedures as documented in enhancement number two (2) contained in LRA Section B.3.1.1, and Section A.5, Item 43. Therefore, the Fatigue Monitoring (B.3.1.1) aging management program will monitor the transient cycles and severities and will require action prior to exceeding a fatigue usage of 1.0.

RAI 4.3.1-3, Transient 16, regarding recovery of main feedwater flow after isolation (060)

Applicability: Byron and Braidwood

Background:

LRA Tables 4.3.1-1 and 4.3.1-5 state that Transient 16, “Recovery of Main Feedwater Flow After Isolation (Unit 1 only),” is applicable to the Unit 1 Steam Generators only for both Byron and Braidwood. The LRA further states that the transient was not evaluated separately because cycles associated with switching between main and auxiliary feedwater flow are implicit in the cycles counted for the other reactor coolant system (RCS) transients.

Issue:

It is unclear to the staff which “other RCS transients” will be monitored since they are implicit in cycles associated with switching between main and auxiliary feedwater flow. The applicant did not provide enough information to describe why monitoring these “other RCS transients” will accurately account for Transient 16. It is also unclear to the staff why Transient 16 is applicable to Byron, Unit 1, and Braidwood, Unit 1, only.

Request:

1. Clarify which RCS transients will be monitored to account for Transient 16. Explain and justify how monitoring these other RCS transients will be adequate so that Transient 16 will not need to be monitored through the period of extended operation.

2. Confirm that these other RCS transients monitored in lieu of Transient 16 are included in applicable LRA tables and will be incorporated into the Fatigue Monitoring Program. If not, revise the LRA to ensure that these other RCS transients are identified in the appropriate LRA tables in LRA Section 4.3 and are included in the Fatigue Monitoring Program.

3. Explain and justify why Transient 16 is only applicable to only Byron, Unit 1, and Braidwood, Unit 1. Clarify if the other RCS transients monitored in lieu of Transient 16 is applicable only to Byron, Unit 1, and Braidwood, Unit 1.

Exelon Response:

LRA Table 4.3.1-1 documents RCS transient baseline cycles and 60-year cycle projections for Byron Units 1 and 2, including transient number 16, “Recovery of Main Feedwater Flow After Isolation (Unit 1 only).” LRA Table 4.3.1-4 (not Table 4.3.1-5) contains transient 16, “Recovery of Main Feedwater Flow After Isolation (Unit 1 only)” for Braidwood Unit 1 and 2.

1. [

]a,c,e

Transient Number from LRA Table 4.3.1-1 and 4.3.1-4	RCS Transient Description	CLB Cycle Limit
20	Loss of Load	80
21	Loss of Power	40
22	Partial Loss of Flow	80
23	Reactor Trip from Full Power: Case A – with no inadvertent cooldown	230
24	Reactor Trip from Full Power: Case B – with cooldown and no safety injection	160
25	Reactor Trip from Full Power: Case C – with cooldown and safety injection	10
26	Inadvertent RCS Depressurization	20
28	Control Rod Drop	80
29	Inadvertent Safety Injection (ECCS) Actuation	60
	Total	760

Descriptions of the Replacement Steam Generator (RSG) design transients in the design basis analysis indicate that [

][a,c,e] If these RCS transients remain within the allowable number of cycles for 60 years, transient 16 will also remain within the allowable number of cycles for 60 years. Therefore, a separate accounting for transient 16 is not necessary.

2. All of these transients are included in LRA Tables 4.3.1-1 and 4.3.1-4. The transients and limits documented in the above table are currently monitored by Fatigue Monitoring (B.3.1.1) aging management program implementing procedures and will continue to be monitored during the period of extended operation.

3. Transient 16 (Recovery of Main Feedwater Flow After Isolation) is only applicable to the Unit 1 Replacement Steam Generators at Byron and Braidwood Stations based on the RSG design transient specifications. This transient is not a part of the design basis or current licensing basis for the Unit 2 steam generators at Byron and Braidwood based on their design transient specifications. The transients listed in the response to Request 1, above, are all applicable to both Units 1 and 2 at Byron and Braidwood.

RAI 4.3.4-1, Use of data to reduce conservatisms in environmentally-assisted fatigue analyses (060)

Applicability: Byron and Braidwood

Background:

LRA Section 4.3.4 states that "where plant specific data was available, it was incorporated into the analysis to reduce conservatism on an as-needed basis for qualification" with respect to environmentally assisted fatigue analyses for the NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," locations.

LRA Section 4.3.4 also states that "since the analyses are based on design cycles and 60-year cycle projections, monitoring of usage through the period of extended operation is required to ensure these conclusions remain valid."

Issue:

The applicant did not identify what plant-specific data was used and the evaluations in which plant-specific data was used to reduce conservatism. It is also not clear in which analysis the applicant has used 60-year projected cycles.

Request:

1. Identify the environmentally assisted fatigue evaluations in which plant-specific data was used. Describe the plant-specific data that was used to reduce conservatism. Provide the basis for the use of the plant-specific data in these environmentally assisted fatigue evaluations.

2. Identify the environmentally assisted fatigue evaluations, including the specific transients and cycles for each location, in which 60-year projected cycles and/or reduced number of cycles were used.

Exelon Response:

1. The environmentally assisted fatigue (EAF) evaluations of the charging nozzles and the pressurizer spray nozzle used plant-specific data to reduce the conservatism of the design transients. All other EAF evaluations used current licensing basis transients.

   The charging nozzle EAF evaluation used plant-specific data to reduce the conservatism of the original "Letdown Flow Shutoff Prompt Return to Service" design transient. This transient is listed in LRA Tables 4.3.1-2 (Byron) and 4.3.1-5 (Braidwood) as Auxiliary System Transient Number 6 in the tables. Plant computer data from Byron and Braidwood Stations, Units 1 and 2, was reviewed for the period of 1999 to 2011. The parameters that were reviewed include regenerative and letdown heat exchanger outlet temperatures, charging and letdown flows, and reactor coolant loop temperatures. Single thermal excursions were characterized by magnitude and rate of temperature change, among other characteristics. Comparison of actual operating transient data against the original design transient demonstrated that operating transients were less severe than the low-cycle design transients. Therefore, it was appropriate to redefine and analyze the transient based on smaller enveloping severities that bounded the operating transients. Review of the plant data from each unit showed a large number of transient cycles associated with flow isolation design transients with temperature changes (ΔT) that were significantly less than the ΔT associated with the design transients but exceeded the design cycles. The thermal cycles corresponding to the "Letdown Flow Shutoff Prompt Return to Service" cycles were therefore reclassified into bounding maximum ΔT ranges of [ ][a,c,e] based on actual transient events occurring on all four units. Cycles of each range were determined from the reduced data and pro-rated over past operation and future operation through 60 years. Extrapolation was justified based on operator interviews and review of each unit's plant operating procedures that would affect charging and letdown flow. Procedures have not changed significantly from initial plant startup to 1999. Therefore, the large amount of plant data that was reviewed from 1999 to 2011 is representative of prior operating years at Byron and Braidwood. These four bounding redefined transients were inputs to the charging nozzle EAF evaluation.

The pressurizer spray nozzle EAF evaluation used plant-specific data to reduce the conservatism of the "Plant Heatup" and "Plant Cooldown" design spray transients. Plant computer data from Byron and Braidwood Stations, Units 1 and 2, was reviewed for the period of 1999 to 2012. The parameters that were reviewed include pressurizer spray line temperature, pressurizer spray line flow demand, pressurizer steam and water temperatures, and reactor coolant loop temperatures. Review of the plant-specific heatup and cooldown spray data showed that the design heatup and cooldown transients for the pressurizer spray nozzle were defined with a conservative number of spray events and spray nozzle ΔT values for Byron and Braidwood Stations, Units 1 and 2. The number of spray events per heatup and cooldown were determined by looking at spray flow demand. The ΔT values were recorded at the times a spray event occurred, and the cycles were classified and counted according to enveloping ΔT ranges. Cycles of each range were determined from the reduced data and pro-rated over past operation and future operation through 60 years. Extrapolation was justified based on Byron and Braidwood operator interviews and review of plant operating procedures that would affect pressurizer spray operation. Procedures have not changed significantly from initial plant startup to 1999. Therefore, the large amount of plant data that was reviewed from 1999 to 2012 is representative of prior operating years at Byron and Braidwood. These bounding redefined transients were inputs to the pressurizer spray nozzle EAF evaluation.

2. The environmentally assisted fatigue (EAF) evaluations of the charging nozzles, accumulator nozzles, safety injection nozzles, and the pressurizer spray nozzle used 60-year projected cycles or a reduced number of cycles (several locations used a number of cycles between design and 60-year projected cycles to provide additional margin). All other EAF evaluations used design transients and cycles.

   The locations in which 60-year projected cycles or a reduced number of cycles were used for the EAF evaluations, along with the specific transients and corresponding cycles, are shown in the following tables. Note that the limiting numbers of cycles for Byron and

Braidwood Stations, Units 1 and 2 were used in the EAF evaluations. As stated in the disposition of the TLAA in LRA Section 4.3.4, the Fatigue Monitoring program will monitor these transient cycles and require action prior to exceeding the environmental fatigue usage limit of 1.0. The reduced cycles used for the EAF evaluations will become the period of extended operation (PEO) CLB Cycle limits.

<u>Charging Nozzles</u>

Note that the charging nozzle EAF evaluation used input cycle values [ ][a,c,e] of the 60-year projected total cycles for auxiliary transients. Procedures require that each charging line (normal and alternate) is only used 50% of the time. Use of [ ][a,c,e] of the 60-year projected total cycles is consistent with the design transient assumption and provides conservatism in the evaluation for each nozzle. Further discussion of the charging nozzle transient projections is contained in response to NRC RAI 4.3.1-1. Baseline cycles were based on a review of plant data and projected based on heatup/cooldown projections for flow shutoff transients and unit load/unload projections for flow change transients.

Auxiliary System Transient Number and Description for Charging Nozzle from LRA Tables 4.3.1-2 and 4.3.1-5	**EAF Evaluation Cycles (PEO CLB Cycle Limits)**
12. Charging Flow Shutoff with Delayed Return to Service (Binned with) 13. Charging and Letdown Flow Shutoff and Return to Service	55
11. Charging Flow Shutoff with Prompt Return to Service	12
7. Charging Flow Step Decrease and Return to Normal	3,500
8. Charging Flow Step Increase and Return to Normal	3,500
5. Letdown Flow Shutoff Delayed Return to Service	12
9. Letdown Flow Step Decrease and Return to Normal	750
10. Letdown Flow Step Increase and Return to Normal	3,500
6. Letdown Flow Shutoff Prompt Return to Service (Note 1)	
Case A: [ ][a,c,e]	70
Case B: [ ][a,c,e]	180

Auxiliary System Transient Number and Description for Charging Nozzle from LRA Tables 4.3.1-2 and 4.3.1-5	**EAF Evaluation Cycles (PEO CLB Cycle Limits)**
Case C: [ ][a,c,e]	15
Case D: [ ][a,c,e]	25

NA- Not applicable
Notes:
1. Explanation of the development of this redefined transient below is contained in the response to Request 1 above and response to NRC RAI 4.3.1-1.

Accumulator Nozzles

RCS Transient Number and Description for Accumulator Nozzles from LRA Tables 4.3.1-1 and 4.3.1-4	**EAF Evaluation Cycles (PEO CLB Cycle Limit)**
26. Inadvertent RCS Depressurization	2
29. Inadvertent Safety Injection (ECCS) Actuation	9
20. Loss of Load	6
21. Loss of Power	6
1. Plant Heatup at 100^{o}F/hr	117
2. Plant Cooldown at 100^{o}F/hr	117
38. Primary Side Hydrostatic Test*	0
40. Primary Side Leak Test	40
15. Refueling	40
Auxiliary System Transient Number and Description for Accumulator Nozzles from LRA Tables 4.3.1-2 and 4.3.1-5	
1. Accumulator Operation	3

*10 cycles of the Primary Side Hydro Test transient were removed per ASME Code Section NB-3226 (e).

Safety Injection Nozzles

RCS Transient Number and Description for Safety Injection Nozzles from LRA Tables 4.3.1-1 and 4.3.1-4	**EAF Evaluation Cycles(PEO CLB Cycle Limits)**
25. Reactor Trip from Full Power: Case C - with cooldown and safety injection	5
26. Inadvertent RCS Depressurization	2
29. Inadvertent Safety Injection (ECCS) Actuation	9

Pressurizer Spray Nozzle

RCS Transient Number and Description for Pressurizer Spray Nozzle from LRA Tables 4.3.1-1 and 4.3.1-4	**EAF Evaluation Cycles(PEO CLB Cycle Limits)**
1. Plant Heatup at 100^{o}F/hr *	117
2. Plant Cooldown at 100^{o}F/hr *	117
5. Unit Loading @ 5% of Full Power/Min**	2,550
6. Unit Unloading @ 5% of Full Power/Min**	2,550
7. 10% Step Load Increase	727
8. 10% Step Load Decrease	727
9. Large Step Load Decrease with Steam Dump	75
12. Boron Concentration Equalization	5,074
20. Loss of Load	10
22. Partial Loss of Flow	10
23. Reactor Trip from Full Power: Case A - with no inadvertent cooldown	100
24. Reactor Trip from Full Power: Case B - with cooldown and no safety injection	50
28. Control Rod Drop	20
29. Inadvertent Safety Injection (ECCS) Actuation	15
Auxiliary System Transient Number and Description for Pressurizer Spray Nozzle from LRA Tables 4.3.1-2 and 4.3.1-5	
21. Pressurizer Spray and Auxiliary Spray Line Piping and Nozzles Transients	6

*Plant Heatup and Plant Cooldown transients for the pressurizer spray nozzle include reduced spray cycles per heatup/cooldown and reduced spray ΔT's as explained in the response to Request 1.

**Load changes from 0-10% in magnitude were removed from the 60-year projected number of cycles because pressurizer spray operations are typically not used during small load changes. The new number of projected cycles applies only to the pressurizer spray nozzle.

RAI 4.3.4-2, WESTEMS™ Metal Fatigue calculation methodology (060)

Applicability: Byron and Braidwood

Background:

LRA Section 4.3.4 states that "the WESTEMS™ fatigue calculation methodology employs a conservative algorithm for selection of the stress peaks and valleys for use in the ASME fatigue evaluation. In some cases, conservatism may be removed by the analyst using optional program tools, to produce a more accurate final result. For any ASME component fatigue evaluation in which the analyst removed conservatism in the peak and valley selection, full documentation of the justification of peak removal was included in the supporting calculations. Otherwise, the conservatism inherent to the WESTEMS™ software was retained for the ASME fatigue evaluations."

Issue:

It is not clear to the staff whether the applicant used these "optional program tools" to remove conservatism to produce a more accurate final result for fatigue evaluations and the applicant's basis for removal of this conservatism.

Request:

1. Clarify whether any of the fatigue evaluation has used these "optional program tools" to remove conservatism.
    a) If so, identify all the fatigue evaluations in which these "optional program tools" were used. Provide three examples in which these "optional program tools were used. For each example, provide the basis for the removal of conservatism and justify that a more accurate final result was produced.

Exelon Response:

1. The fatigue evaluations of the Reactor Coolant Loop (RCL) Safety Injection Nozzles, Reactor Coolant Loop Accumulator Nozzles, and the Pressurizer Spray Nozzle involved the use of peak editing tools (e.g., optional program tools) in the WESTEMS™ software peak and valley stress state selection. All other fatigue evaluations did not use peak editing tools and the original peak and valley stress states that were selected by the WESTEMS™ software peak and valley algorithm were conservatively retained in the fatigue analysis.

    In each of the three evaluations above, the removed peaks were determined to be non-controlling and redundant, which resulted in unnecessary conservatism. These determinations were made using the WESTEMS™ User Manual procedures with full documentation included in the supporting calculations as discussed in the LRA Section 4.3.4, page 4.3-28.

a) One example from each of the three evaluations above, in which peak editing tools were applied to remove conservatism in stress peaks for a transient, is provided below:

Reactor Coolant Loop Safety Injection Nozzles

Figure 1 shows a plot of Total stress intensity and Primary plus Secondary stress intensity for a Reactor Coolant Loop Safety Injection Nozzle transient used in the fatigue evaluation. [

]a,c,e Even though the curve shows a total stress peak greater than the primary plus secondary stress peak, the total stress peak at 41 seconds contributed less to the alternating stresses in the initial fatigue evaluation than the Primary plus Secondary stress peak at 73 seconds and was determined to be non-controlling and redundant. Therefore, based on the criteria in the WESTEMS™ User's Manual, the peak time at 41 seconds for this transient was removed from the fatigue restart file and the fatigue calculation was re-run. Due to consideration of the controlling peak times, the analysis resulted in the removal of unnecessary conservatism and more accurate usage factors.

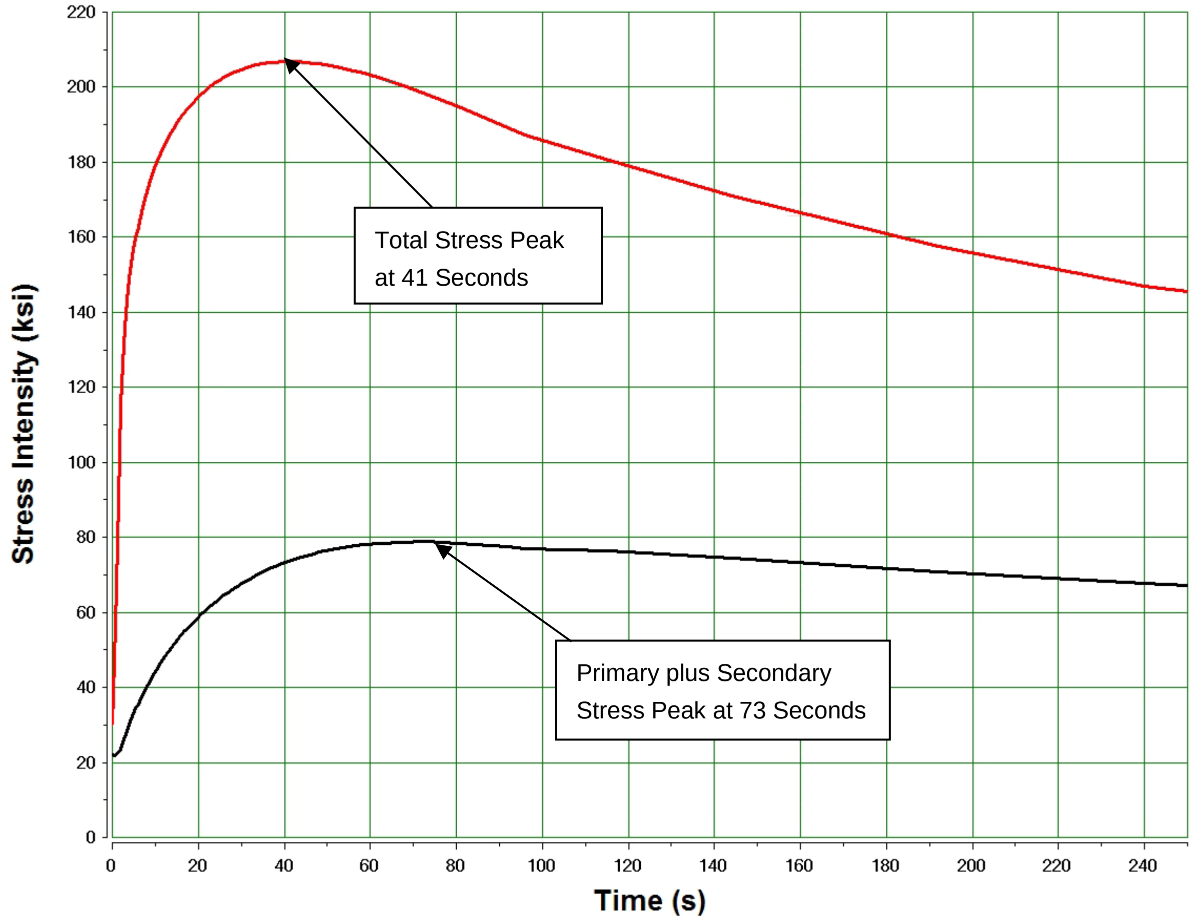

Figure 1: Total and Primary Plus Secondary Stress Intensity Peaks for a RCL Safety Injection Nozzle Transient

Reactor Coolant Loop Accumulator Nozzles

Figure 2 shows a plot of Total stress intensity and Primary plus Secondary stress intensity for a Reactor Coolant Loop Accumulator Nozzle transient used in the fatigue evaluation. [

][a,c,e] The Primary plus Secondary stress peak at 15 seconds contributed less to the alternating stresses in the initial fatigue evaluation than the Total stress peak at 9 seconds and was determined to be non-controlling and redundant. Therefore, based on the criteria in the WESTEMS™ User's Manual, the peak time at 15 seconds for this transient was removed from the fatigue restart file and the fatigue calculation was re-run. Due to consideration of the controlling peak times, the analysis resulted in the removal of unnecessary conservatism and more accurate usage factors.

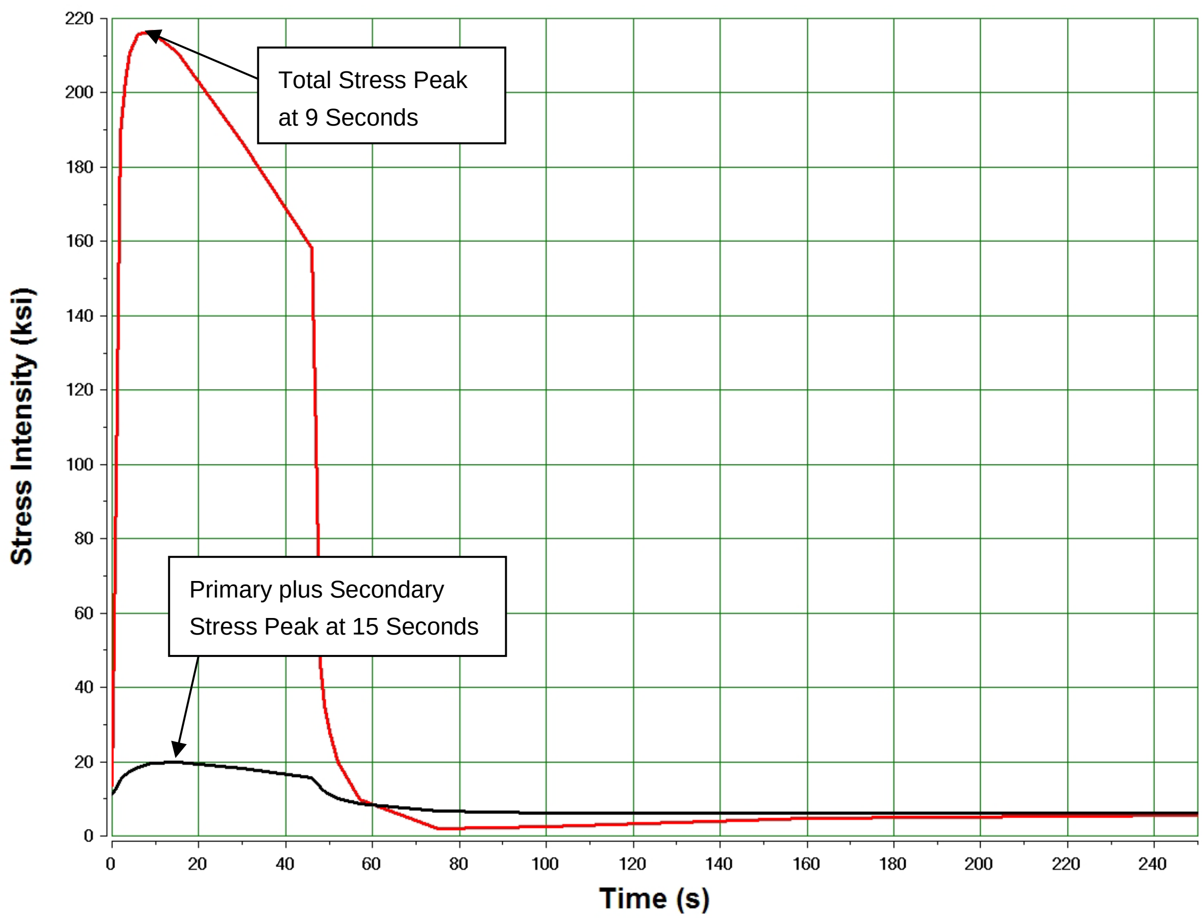

Figure 2: Total and Primary Plus Secondary Stress Intensity Peaks for a RCL Accumulator Nozzle Transient

Pressurizer Spray Nozzle

Figure 3 shows a plot of Total stress intensity and Primary plus Secondary stress intensity for a Pressurizer Spray Nozzle transient used in the fatigue evaluation. [

]a,c,e The Primary plus Secondary stress peak at 632 seconds contributed less to the alternating stresses in the initial fatigue evaluation than the Total stress peak at 620 seconds and was determined to be non-controlling and redundant. Therefore, based on the criteria in the WESTEMS™ User's Manual, the Primary plus Secondary stress peak at 632 seconds for this transient was removed from the fatigue restart file and the fatigue calculation was re-run. Due to consideration of the controlling peak times, the analysis resulted in the removal of unnecessary conservatism and more accurate usage factors.

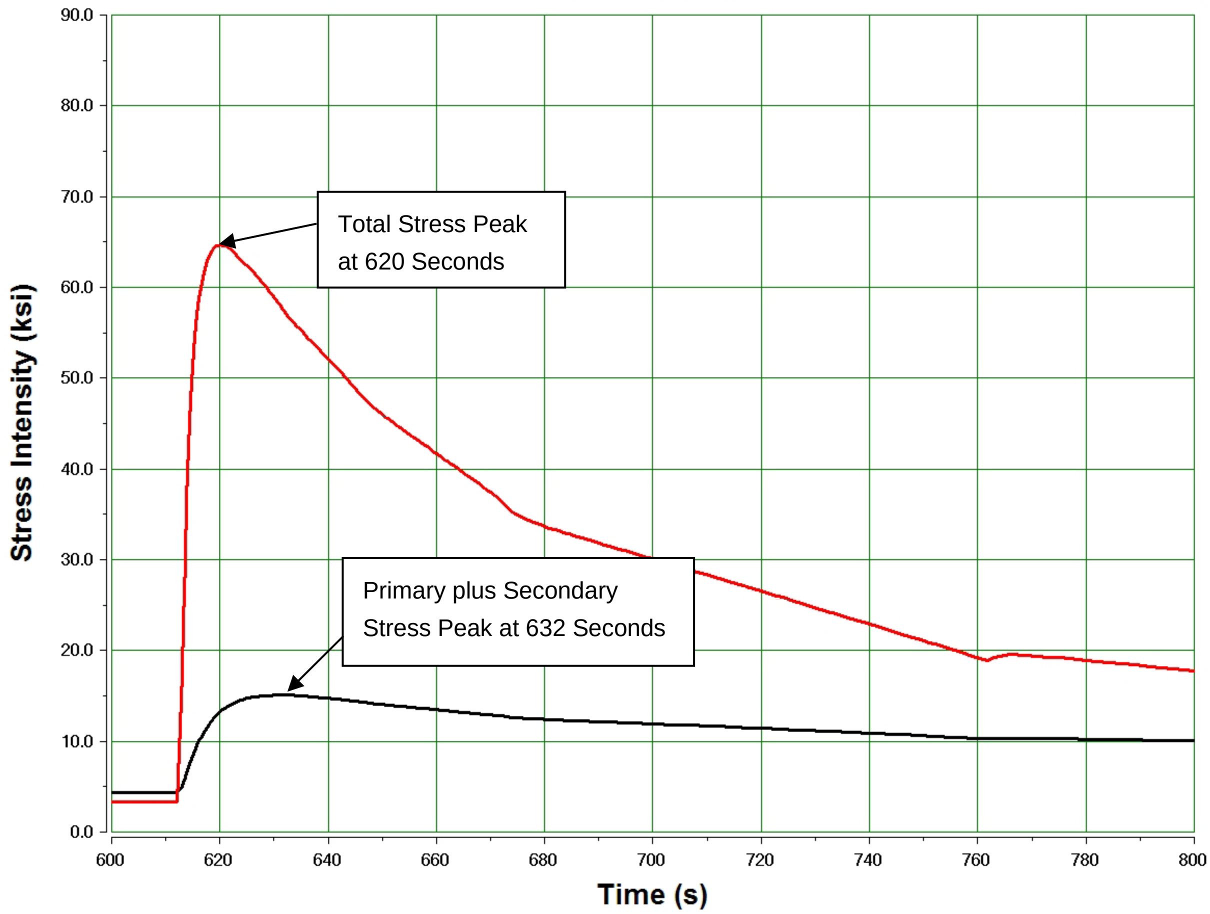

Figure 3: Total and Primary Plus Secondary Stress Intensity Peaks for a Pressurizer Spray Nozzle Transient

RAI 4.3.4-3, Determining most limiting cumulative usage factor for metal fatigue transient sections (060)

Applicability: Byron and Braidwood

Background:

LRA Section 4.3.4 states that the Class 1 components were grouped into transient sections, which is defined as a group of sub-components/locations that experience the same transients. The LRA further states that components that reside in the same transient section can easily be compared with each other to determine the most limiting component (or leading location) which is the location with the highest cumulative usage factor (CUF) value. The differences in stresses experienced by each component in a transient section are generally the result of the material and geometry differences.

Issue:

The staff noted that in order to have a meaningful comparison of CUF values to determine the most limiting component (or leading location) by using the highest CUF value, it is important that the CUFs were assessed similarly (e.g., amount of rigor in calculating CUF) and used the same fatigue curves in ASME Code, Section III, Appendix I. It is also not clear whether the applicant considered the differences in component materials when comparing CUF values since material properties may impact the CUF values. The staff noted that through the course of plant operation it is possible that CUF values for specific components were possibly re-evaluated as part of power uprates, generic letters, bulletins, etc. to different editions of ASME Code, Section III and with varying levels of rigor when compared to the fatigue evaluations performed for the plant's original design.

Request:

1. Confirm that the CUFs that were compared with each other in a transient section to identify the location with the highest CUF value were assessed similarly (e.g., amount of rigor in calculating CUF) and used the same fatigue curves in ASME Code Section III Appendix I to provide a meaningful comparison. If not, provide the basis for ranking or comparing the CUFs to one another to provide an appropriate method for screening and determining a leading/limiting location.

2. Clarify whether CUF values of different material types were compared to one another when determining the leading location(s) within a transient section. If yes, identify the transient section, locations and materials that have been compared and eliminated for consideration of EAF. Justify that this comparison of CUF values between different materials within a transient section for the consideration of EAF is appropriate or valid.

Exelon Response:

Refer to RAI 4.3.4-5, response to Request 2, for additional information concerning the screening methodology used to determine limiting locations for environmentally assisted fatigue.

1. Locations within a transient section were compared similarly, considering the amount of rigor used in calculating the CUF. The basis for ranking or comparing the CUF's to one another appropriate for screening locations within a transient section, and determining a leading/limiting location, is detailed in the response to RAI 4.3.4-5, Request 2.

   As part of the consistent stress analysis method basis comparisons, all locations with materials other than nickel alloy that were compared within a transient section used the same fatigue curves from the ASME Code Section III Appendix I for each respective material. For nickel alloy locations, the effect of the NUREG/CR-6909 fatigue curve was considered when comparing to other locations in a transient section.

   The results of the EAF screening of components associated with equipment are presented in LRA Table 4.3.4-2. The results of the EAF screening of piping components are presented in LRA Table 4.3.4-3.

2. The Byron and Braidwood Stations Class 1 piping components are all stainless steel. Therefore, within a transient section associated with Class 1 piping components, all materials were the same.

   The EAF screening evaluation for the equipment locations considered different materials within a transient section as applicable to Byron and Braidwood Stations, Units 1 and 2. It is necessary, and appropriate, to consider different material types in the comparison within a transient section, since the material type influences the applicable F_{en} value and hence, a given location's ranking in the EAF screening process. For example, the transient section in the Reactor Vessel outlet nozzle region consists of stainless steel (safe end), low alloy steel (nozzle), and nickel alloy (safe end to nozzle weld) materials. As explained in the response to RAI 4.3.4-5, Request 2, screening F_{en} penalty factors were calculated for each location in a transient section based on material. Therefore, for the Reactor Vessel outlet nozzle transient section F_{en} penalty factors were calculated for the stainless steel locations using NUREG/CR-5704, the low alloy steel locations using NUREG/CR-6583, and the nickel alloy locations using NUREG/CR-6909. The Material F_{en} penalty factors were applied to each respective CUF to determine a Material CUF_{en}. The nickel alloy locations were also addressed with respect to the NUREG/CR-6909 fatigue curve impact on the comparison process. All three locations were evaluated with the same analytical method and rigor. The safe end location produced the highest screening CUF_{en} greater than 1.0 for this transient section. Therefore, the safe end location was selected as the leading location for the Reactor Vessel outlet nozzle transient section. In addition to this example, the response to RAI 4.3.4-6 describes similar examples for consideration of different material types in performing a consistent comparison in the EAF screening process. These are considered to be typical and bounding examples of how CUF values of different material types were compared to one another when determining the leading location(s) within a transient section for equipment. Similar to the responses for requests in RAI's 4.3.4-4 and 4.3.4-5, these examples are a sample set of locations that are bounding for the population of eliminated locations, since they include consideration of equipment locations with higher design CUF values and the various material differences that influence both fatigue usage and F_{en}.

Based on the method utilized to determine the Material CUF_{en} values within a transient section as described above, leading or limiting locations were appropriately determined considering the comparison of different materials CUF values during the EAF screening process.

RAI 4.3.4-4, Determinations of environmentally-assisted fatigue related to components in various sections but in the same major component or system (060)

Applicability: Byron and Braidwood Stations

Background:

LRA Section 4.3.4 states that components that reside in different transient sections, but are within a common system or piece of major equipment, were also compared to determine leading locations to represent their respective system/equipment.

The LRA also states that often, it is the transients themselves that control which components have the highest usage factors in a given system, and so, within a particular system, those transient sections with the most severe system transients will usually have the components with the highest usage factors. However, the applicant stated that the comparison of components in different transient sections must be performed after the appropriate environmental fatigue correction factor (F_{en}) is applied to the component usage factor because F_{en} is dependent on temperature and strain rate and, therefore, can vary for each transient section.

Issue:

Based on the information in the LRA, it is not clear when the applicant compared components that reside in different transient sections, but are within a common system or piece of major equipment, to determine leading locations to represent their respective system/equipment. It is also not clear what assumptions or factors were considered by the applicant when making this comparison to determine the leading location that resides in different transient sections and the basis for eliminating a location for consideration of EAF.

Request:

1. Identify the locations that were compared from different transient section, but within a common system or piece of major equipment, and the component that was eliminated. A sample set of locations and eliminated component(s) may be provided: however, a justification of sufficient detail is necessary to explain that this sample set is bounding for the population of compared locations and eliminated component(s).

2. For each of these situations, provide the basis for the comparison that was made for different transient sections but within a common system or piece of major equipment. In addition, provide the basis for eliminating the component(s) that was eliminated for consideration of EAF. As part of these justifications, specifically address any assumptions, factors or criteria that were applicable when implementing this comparison.

Exelon Response:

Refer to RAI 4.3.4-5, response to Request 2, for additional information concerning the screening methodology used to determine limiting locations for environmentally assisted fatigue.

1. Based on the process described in the response to Request 2, all locations within each piping system or major equipment with a fatigue analysis were addressed and compared, and all but the identified leading locations were justified to be eliminated. An example of the comparison between transient sections as a result of applying the process for one piping system is provided here. The example is for the cold leg safety injection accumulator piping transient section 1 (as shown in Figure 1 below), which extends from the reactor coolant loop (RCL) cold leg to the first check valve, and transient section 2, which extends from the first check valve to the safety injection branch line. This system is considered as a bounding example since the safety injection nozzle (RCL Nozzle) is a NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components” location. In addition, these sections are bounding representations for this system, since the transients and components in transient sections 1 and 2 are similar or more limiting than those in transient sections 3, 4, and 5. Figure 1 illustrates the system transient sections and the accompanying Table 1 below illustrates the result of using the EAF screening process and determination of a leading location when comparing two transient sections.

Figure 1 - Example of Cold Leg Safety Injection Accumulator Piping Components and Respective Transient Sections

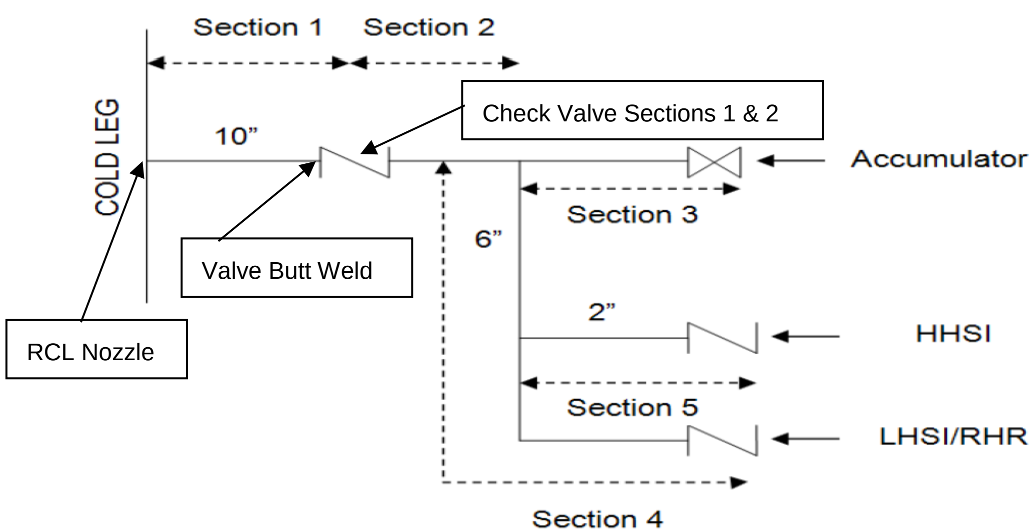

Table 1

Example of Cold Leg Safety Injection Accumulator Piping Components and Respective Transient Sections

Component	Transient Section	EAF Screening Process Results
RCL Nozzle	1	Retained
Check Valve	1	Eliminated by stress basis comparison
Valve Butt Weld	1	Eliminated by stress basis comparison
Check Valve	2	Eliminated by stress basis comparison

2. The general EAF screening process is detailed in the response to RAI 4.3.4-5 Request 2.

   To compare locations from different transient sections but within a common system or piece of major equipment, the leading locations from each transient section within a common system or piece of major equipment needed to be determined. An example of this process being applied to transient sections 1 and 2 of the cold leg safety injection accumulator piping is detailed below.

Step 1 - Data Collection

All of the pertinent inputs were collected from all the Byron and Braidwood units. This included the component materials, drawings, and current licensing basis fatigue evaluations (and fatigue curves) for the cold leg safety injection accumulator piping system.

Step 2 - Transient Section Considerations

The transient sections for the cold leg safety injection accumulator piping system were determined in the current licensing basis fatigue evaluation. See Figure 1 of Response 1.

Step 3 - Screening F_{en} Calculation

i. Maximum environmental fatigue correction factors (F_{en}) were calculated for each component within a transient section based on material (Material F_{en}).

   Several locations from transient section 1 of the piping system are compared in the table below. All three (3) components are stainless steel material. The maximum Material F_{en} was calculated as described in the general EAF screening process.

Transient Section	Component	CUF	Material F_{en}	Material CUF_{en}
1	RCL Nozzle	0.95	15.35	14.58
1	Check Valve	0.48	15.35	7.37
1	Valve Butt Weld	0.5368	15.35	8.24

Since all three (3) components are the same material, the maximum F_{en} penalty factor for each using these assumptions is the same. Each of the locations has a Material CUF_{en} greater than 1.0. Therefore, all three (3) locations are retained for this step.

ii. Maximum environmental fatigue correction factors (F_{en}) were calculated for each component within a transient section based on material and maximum transient section temperature (Temperature F_{en}).

The locations from transient section 1 of the piping system are compared in the table below. Because all three (3) components are in transient section 1, the same temperature applies. The maximum temperature F_{en} was calculated as described in the general EAF screening process.

Transient Section	Component	CUF	Maximum Temperature	Temperature F_{en}	Temperature CUF_{en}
1	RCL Nozzle	0.95	>392°F	15.35	14.58
1	Check Valve	0.48	>392°F	15.35	7.37
1	Valve Butt Weld	0.5368	>392°F	15.35	8.24

Since the same maximum temperature applies to all three (3) components, the maximum F_{en} penalty factor for each using these assumptions is the same. Each of the locations has a Temperature CUF_{en} greater than 1.0. Therefore, all three (3) locations are retained for this step.

Step 4 - Stress Basis Comparison

Because all three (3) locations in transient section 1 remain candidate locations after step 3, it was necessary to perform a stress basis comparison. The results are shown in the table below.

The transient section 1 RCL nozzle was qualified to ASME Section III, NB-3600 but using Finite Element Analysis for thermal and mechanical stress quantities (analysis ranking = 4). The transient section 1 check valve was qualified to ASME Section III, NB-3545, which is a conservative equation-based methodology to combine stresses for fatigue, similar to that of ASME Section III, NB-3600 (analysis ranking = 1). The transient section 1 valve butt weld was qualified to ASME Section III, NB-3600 but using Finite Element Analysis only for thermal stress quantities (analysis ranking = 3).

Transient Section	Component	CUF	Final Screening F_{en}	Final Screening CUF_{en}	Stress Analysis Method Ranking
1	RCL Nozzle	0.95	15.35	14.58	4
1	Valve Butt Weld	0.5368	15.35	8.24	3
1	Check Valve	0.48	15.35	7.37	1

The final screening CUF_{en} for the transient section 1 check valve is less than that of the transient section 1 valve butt weld, and the transient section 1 check valve was qualified using a less rigorous analysis methodology than the transient section 1 valve butt weld. Therefore, it is concluded that the transient section 1 valve butt weld is more limiting than the transient section 1 check valve.

The final screening CUF_{en} for the transient section 1 valve butt weld is less than that of the transient section 1 RCL Nozzle, and the transient section 1 valve butt weld was qualified using a less rigorous analysis methodology than the transient section 1 RCL Nozzle. Therefore, it is concluded that the transient section 1 RCL Nozzle is more limiting than the transient section 1 valve butt weld.

Step 5 - Leading Location Identification

The leading location for transient section 1 of the cold leg safety injection accumulator piping system is the RCL Nozzle as explained in step 4. The same process was applied to transient section 2 of the cold leg safety injection accumulator piping system. The leading location for transient section 2 was determined to be the check valve.

The next step compared the leading locations from transient sections 1 and 2. The results are shown in the table below.

The transient section 2 check valve was qualified to ASME Section III, NB-3545, which is a conservative equation-based methodology to combine stresses for fatigue, similar to that of ASME Section III, NB-3600 (analysis ranking = 1). It should be noted the final screening F_{en} is different for the check valve when computing the final screening CUF_{en} in transient section 2 due to the difference in the maximum transient section temperature.

Transient Section	Component	CUF	Final Screening F_{en}	Final Screening CUF_{en}	Analysis Ranking
1	RCL Nozzle	0.95	15.35	14.58	4
2	Check Valve	0.48	2.547	1.22	1

The final screening CUF_{en} for the transient section 2 check valve is significantly less than that of the transient section 1 RCL Nozzle, and the transient section 2 check valve was qualified using a less rigorous analysis methodology than the section 1 RCL Nozzle. Therefore, it is concluded that the transient section 1 RCL Nozzle is more limiting than the transient section 2 check valve.

RAI 4.3.4-5, Use of stress based comparisons to remove components or locations from EAF consideration (060)

Applicability: Byron and Braidwood Stations

Background:

LRA Section 4.3.4 states that a stress basis comparison is performed to identify the leading transient section locations. The LRA states that Westinghouse has developed an approach that was applied to Byron and Braidwood, Units 1 and 2, for performing a stress basis comparison for the components included in the screening process.

The applicant stated that the following stress analysis characteristics were considered in determining the limiting locations within a given transient section:

1) Qualification Criteria (ASME Code Section III, NB-3200, NB-3600, etc.)
2) Stress Analysis Technique

Furthermore, the applicant stated that in order to perform these stress basis comparisons, a hierarchy of stress analysis techniques was developed based on fatigue analysis experience to define the relative complexity of the various techniques.

1) Standard NB-3600 analysis
2) NB-3600 with non-standard mechanical stress indices or stress quantities used in stress formulas
3) NB-3600 with non-standard thermal stress indices or stress quantities used in stress formulas
4) Combination of 2) and 3)
5) NB-3200 Fatigue Analysis

Those components with a screening environmentally-adjusted cumulative usage factor (CUF_{en}) of less than 1.0 were removed from the list because they have been calculated using the design basis fatigue usage factors with a maximum F_{en} based on material.

Issue:

The staff noted that the stress basis comparison described in LRA Section 4.3.4 consists of two aspects: (1) consideration of stress analysis characteristics; and (2) a hierarchy of stress analysis techniques.

The staff noted that it appears the applicant eliminated certain Safety Class 1 reactor pressure boundary locations susceptible to EAF by performing a “stress basis comparison.” It is not clear which locations were eliminated or what the technical basis was for removing these locations from consideration of EAF as a leading location.

<u>Request</u>:

1. Confirm whether the use of a stress basis comparison and screening CUF_{en} of less than 1.0 were the only methods for eliminating locations for consideration of EAF. If not, describe and justify any other methods that were used. Include the locations that were eliminated and the associated technical basis.

2. Describe and justify the circumstances and situation when locations were eliminated using a stress basis comparison.

3. Identify the locations that were eliminated as a result of performing this stress basis comparison and provide the basis for eliminating these locations/components. Specifically, address any assumptions, factors, or criteria that were used when eliminating these locations for consideration for EAF. A sample set of locations may be provided; however, a justification of sufficient detail is necessary to explain that this sample set is bounding for the population of eliminated locations.

<u>Exelon Response:</u>

1. The use of a stress basis comparison for locations with a screening CUF_{en} greater than or equal to 1.0; and a screening CUF_{en} of less than 1.0, were the only methods for eliminating locations for consideration of Environmentally Assisted Fatigue (EAF). Per Request 3, a sample set of locations eliminated are provided in the response below.

2. The general EAF screening process is detailed in the steps below and includes the circumstances and situations for elimination of locations using stress basis comparison and the screening CUF_{en}:

   <u>Step 1 - Data Collection</u>

   All of the pertinent inputs were collected from Byron and Braidwood. This included all of the Safety Class 1 reactor coolant pressure boundary component materials, drawings, and current licensing basis fatigue evaluations. Current licensing basis fatigue curves were identified.

   <u>Step 2 - Transient Section Considerations</u>

   The transient sections were determined for all piping systems and major equipment (reactor vessel, pressurizer, etc.) included in the screening evaluation. A transient section is defined as a group of sub-components or locations that experience the same transients. Components within each transient section were evaluated initially as a group before they were compared against other components within the same system/equipment. Components within different transient sections but within a common system or major equipment were also compared later in the process in step 5, iii below. The comparison of components in different transient sections must be done after the appropriate environmental fatigue correction factor (F_{en}) is applied to the component fatigue usage (CUF), because the F_{en} correction factor is dependent on temperature and strain rate and therefore can vary for each transient.

Step 3 - Screening F_{en} Calculation

i. Maximum/bounding environmental fatigue correction factors (F_{en}) were calculated for each component within a transient section based on material (Material F_{en}).
To calculate the maximum Material F_{en}, bounding assumptions were made regarding the parameters (temperature, dissolved oxygen, sulfur content, strain rate) used in the F_{en} equations from the NUREG reports (NUREG/CR-6583, NUREG/CR-5704, and NUREG/CR-6909). Generally, parameters were chosen such that the maximum F_{en} penalty factor for the material would be calculated. The only exception is dissolved oxygen content. A value of 0.005 ppm was used for the dissolved oxygen (DO) content, which is typical of the PWR environment. For PWR's, except for short periods during heatup/cooldown operations, the DO content is generally well below the 0.05 ppm criteria used to determine the transformed dissolved oxygen content parameter (O*) in the carbon and low alloy steel equations. A review of plant chemistry data from 2004 to 2012 supports the use of 0.005 ppm during normal operation and 0.05 ppm during heatup and cooldown operations. However, elevated DO content usually only occurs when reactor coolant temperature is low. During these periods of operations, fluid temperatures are in the range where the transformed metal temperature parameter (T*) = 0, and the applicable term in the applicable F_{en} equation exponent still reduces to zero. Therefore, in any case for PWR conditions, the term containing the O*T* product is zero in the Fen equations for carbon and low alloy steels. In the stainless and nickel alloy steel equations, use of PWR DO conditions is bounding.

The Material F_{en} was applied to the current licensing basis (CLB) CUF to calculate a screening cumulative usage with EAF for each component (Material CUF_{en}). If required, CLB fatigue curve adjustments were made for nickel alloy materials using NUREG/CR-6909, "Effect of the LWR Coolant Environments on the Fatigue Life of Reactor Materials", to determine the CUF for which the Material Fen was applied in the screening process. Components with a Material CUF_{en} less than 1.0 were eliminated from consideration at this point. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower CUF_{en} than that obtained using the bounding maximum material penalty, and therefore, would remain below 1.0. Eliminating such locations allows the screening comparisons to focus on the remaining locations that are more limiting.

ii. For the locations still remaining in the transient section, refined estimated F_{en} penalty factors were calculated for each component within a transient section based on temperature in an effort to reduce the CUF_{en} to a value below 1.0 (Temperature F_{en} and Temperature CUF_{en}). The maximum Temperature F_{en} was calculated using the same assumptions as the maximum Material F_{en}, except that the maximum temperature for each transient section was input to the applicable

NUREG equation. Components with a Temperature CUF_{en} less than 1.0 were eliminated from consideration at this point. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower CUF_{en} than that obtained using the bounding maximum temperature penalty; and therefore, would remain below 1.0. Eliminating such locations allows the screening comparisons to focus on the remaining locations that are more limiting.

iii. If the Temperature CUF_{en} was greater than 1.0, this location was retained for the next step. The Temperature CUF_{en} is the final screening CUF_{en}.

Step 4 - Stress Basis Comparison

i. For the locations with Temperature CUF_{en} greater than 1.0 in a transient section, the level of technical rigor in the stress analysis method used to calculate the CLB CUF was determined to provide additional comparison.

ii. The most limiting components in each transient section were qualitatively determined using the final screening CUF_{en}s (i.e., the Temperature CUF_{en}) previously calculated and a stress analysis method ranking basis for comparison. The stress analysis method ranking levels are listed from least to most rigorous below:

(1) Standard NB-3600 analysis

(2) NB-3600 with non-standard mechanical stress indices or stress quantities used in stress formulas

(3) NB-3600 with non-standard thermal stress indices or stress quantities used in stress formulas

(4) Combination of (2) and (3) above

(5) NB-3200 Fatigue Analysis:

a) NB-3200 with interaction analysis

b) NB-3200 with elastic Finite Element analysis

c) NB-3228 Elastic/Plastic analysis

Note that this hierarchical list was used primarily for the stress basis comparison of the piping components. Since the majority of the components associated with the equipment are performed using NB-3200 analysis, the components within each piece of equipment were compared within ranking level 5 using an independent ranking system based on the amount of conservatism in the analysis.

In executing the stress basis comparison, elimination of the location with the lower final screening CUF_{en} value and stress analysis method ranking is justified.

This is based on the premise that, if it were analyzed with the same technical rigor as the retained location, its CUF would be even lower, and result in an even lower CUF_{en}.

For situations in which it was difficult to qualitatively determine the most limiting location in a transient section, multiple locations were retained as leading locations.

Step 5 - Leading Location Identification

i. Those components with a Material CUF_{en} or Temperature CUF_{en} less than 1.0 were removed from the final leading location list.

ii. The final screening CUF_{en} and stress basis comparisons were used to identify the leading location within each transient section. Locations with lower screening CUF_{en} and lower analysis method rank were eliminated for the transient section.

iii. Leading locations from each transient section within a piping system or piece of equipment were compared using the final screening CUF_{en}s and stress basis comparisons. In the comparison, the screening CUF_{en} values account for the effects of different materials, transients, and maximum transient section temperatures. Using the stress basis comparisons, elimination of the location with the lower final screening CUF_{en} value and stress analysis method ranking is justified. This is based on the premise that, if it were analyzed with the same technical rigor as the retained location, its CLB CUF would be even lower, and result in an even lower CUF_{en}. Generally, one or two leading locations were able to be identified for each system or piece of equipment.

iv. The system or equipment leading locations were compared against any NUREG/CR-6260 locations within the system or equipment to determine if the NUREG/CR-6260 location was bounding. This included additional stress basis comparisons as required, to ensure a consistent basis of comparison of the leading locations with the NUREG/CR-6260 locations.

A stress basis comparison was only performed if necessary when determining the leading locations, to provide consistent comparison of screening CUF_{en}. For example, if multiple locations remained in a transient section after step 3 in the EAF screening process above, a stress basis comparison was required to determine the amount of technical rigor used in the stress analysis for calculating the CLB CUF. A stress basis comparison was performed as described in step 4 above, to provide a consistent comparison of maximum screening CUF_{en} values within a transient section. A component within a transient section could be eliminated only if its screening CUF_{en} value and stress analysis method ranking were lower than another component being retained. Additional stress basis comparison was performed in step 5 when identifying leading locations for a single system or piece of equipment.

3. The charging lines provide examples of the application of the screening process, and are considered bounding representative locations. This is justified since they are subject to some of the most severe transient loads in the Safety Class 1 systems, typically exhibit significant design fatigue usage in the system components, and experience relatively high temperatures that can result in higher F_{en} values. This is supported by the fact that the charging nozzle is a NUREG/CR-6260 location. An example of the application of a stress basis comparison in a transient section occurred in the normal and alternate charging line piping, specifically for the Reactor Coolant Loop (RCL) nozzle and a pipe butt weld. Both components are stainless steel and both components are located in the same transient section, designated as transient section 1. Transient section 1 extends from the reactor coolant loop cold leg to the first valve. The transient section was selected based on grouping the components experiencing the same thermal transients within the system. The calculated Material CUF_{en} and Temperature CUF_{en} for both components were greater than 1.0. Therefore, both locations were retained for transient section 1 after step 3 of the EAF screening process. To perform a valid comparison of the CUF_{en}, the stress basis comparison was required as described in step 4 of the process above. The RCL nozzle final screening CUF_{en} was 14.28. It was evaluated using NB-3600 with thermal and mechanical finite element stresses and, therefore, has a stress analysis method ranking of 4. The piping butt weld final screening CUF_{en} was 1.36. It was evaluated using a standard NB-3600 analysis, and therefore has a stress analysis method ranking of 1.

   The final screening CUF_{en} for the piping butt weld was significantly less than that of the RCL nozzle and the piping butt weld stress analysis method ranking was lower, which indicates that a more rigorous analysis of the piping butt weld would result in an even lower CUF_{en} than the RCL nozzle. Therefore, it was concluded that the RCL nozzle was more limiting than the piping butt weld, and the piping butt weld was eliminated from further consideration.

   An example of the application of a stress basis comparison between two transient sections in the same system also occurred in the normal and alternate charging line piping. The specific comparison was between the RCL nozzle in transient section 1 described previously and a valve butt weld in transient section 2, which extends from the first valve to the second valve. Both components were found to be the leading location for their respective transient sections, hence the calculated Material CUF_{en} and Temperature CUF_{en} for both components were greater than 1.0. To perform a valid comparison of the CUF_{en}, the amount of technical rigor in the stress analysis method used to calculate each CLB CUF was required in step 5, iii of the process. The RCL nozzle final screening CUF_{en} of 14.28 was evaluated using NB-3600 with thermal and mechanical finite element stresses, and therefore has a stress analysis method ranking of 4. The valve butt weld final screening CUF_{en} of 7.68 was evaluated using NB-3600 with only thermal finite element stress and, therefore, has a stress analysis method ranking of 3.

   The final screening CUF_{en} for the valve butt weld was less than that of the RCL nozzle, and the valve butt weld stress analysis method ranking was lower, which indicates that a more rigorous analysis of the valve butt weld would result in an even lower CUF_{en} than the nozzle. Therefore, it was concluded that the RCL nozzle was more limiting than the valve butt weld, and the valve butt weld was eliminated from further consideration.

   The presentation of the implementation of the EAF screening process using the stress basis comparison for the sample set of locations above provides sufficient detail to bound the population of eliminated locations. Assumptions, factors, and criteria that were used when eliminating these locations for consideration for EAF have been provided. This sample set provides assurance that the locations that were eliminated as a result of performing the EAF screening process using the stress basis comparison were appropriate and justified; and the remaining locations are the leading locations for further consideration of EAF.

RAI 4.3.4-6, Use of NUREG/CR-6909 and its corresponding fatigue curves in calculating cumulative usage factor values (060)

Applicability: Byron and Braidwood Stations

Background:

LRA Section 4.3.4 states that when performing an EAF evaluation, a plant can either use guidance from NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," for austenitic stainless steels, NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," for carbon and low alloy steels, and NUREG/CR-6909, "Effects of the LWR Coolant Environments on the Fatigue Life of Reactor Materials," for nickel alloy steels, or they can use guidance from NUREG/CR-6909 for all materials. Note that if NUREG/CR-6909 is used, the corresponding fatigue curves therein must be considered in calculating the CUF values. This difference must be addressed as part of the EAF screening process. The applicant also indicated that NUREG/CR-6909 was used for nickel alloy steels only.

Issue:

As noted in the LRA, when NUREG/CR-6909 is used, the corresponding fatigue curves in the report must be considered in calculating the CUF values. However, it is not clear to the staff how many/if any nickel alloy steel components were eliminated based on the environmentally-adjusted cumulative usage factor (CUF_{en}) screening process described in the LRA or how the applicant accounted for the difference in fatigue curves used in the fatigue analyses and NUREG/CR-6909 as part of the EAF screening process.

Request:

1. Identify the locations that were eliminated by the CUF_{en} screening process, including the CUF and environmental fatigue correction factor (F_{en}) values for these components. A sample set of locations may be provided; however, a justification of sufficient detail is necessary to explain that this sample set is bounding for population of eliminated locations.

2. Discuss and justify how the difference in fatigue curves used in the fatigue analyses of these components and NUREG/CR-6909 was addressed as part of the EAF screening process.

3. Provide the design basis CUF and revised CUF as a result of the fatigue curves in NUREG/CR-6909.

Exelon Response:

1. The following nickel alloy locations were eliminated by the CUF_{en} screening process. The locations are listed along with each respective design basis CUF and environmental fatigue correction factor (F_{en}).

Component	Design Basis* CUF	F_{en}
Units 1 and 2 Reactor Vessel Outlet Nozzle to Safe End Weld	0.11	4.524
Units 1 and 2 Reactor Vessel Inlet Nozzle to Safe End Weld	0.04	4.524
Unit 1 Replacement Steam Generator (RSG) Tubes	0.19	4.524
Unit 2 Original Steam Generator (OSG) Tubes	0.233	4.524
Unit 2 OSG Tube to Tubesheet Weld	0.379	4.524
Unit 2 OSG Divider Plate (Drain Hole)	0.194	4.524
Unit 2 OSG Divider Plate (Fillet Weld)	0.193	4.524

* Design basis CUF values do not include the effect of NUREG/CR-6909 fatigue curves

2. The NUREG/CR-6909 fatigue curve applicable to nickel alloy steel yields higher incremental CUF's in the high cycle regime, but lower incremental CUF's in the low cycle regime, compared to the ASME Code stainless steel curve used in the CLB analyses. If stress ranges for the controlling fatigue usage pairs from the design fatigue evaluations represent cycles in the low cycle regime of the NUREG/CR-6909 fatigue curve, the contribution from the NUREG/CR-6909 fatigue curve differences would be negligible when an F_{en} penalty factor of 4.524 (maximum nickel alloy penalty factor from NUREG/CR-6909) is applied. However, if the stress ranges for the controlling fatigue usage pairs from the design fatigue evaluations represent cycles in the high cycle regime of the NUREG/CR-6909 fatigue curve, further investigation is required to determine the impact of the NUREG/CR-6909 fatigue curve.

   The reactor vessel outlet and inlet nozzle to safe end welds were evaluated in the CLB using the ASME Code stainless steel fatigue curves. Therefore, the impact of the NUREG/CR-6909 curve was evaluated by comparing the allowable cycles for the CLB analysis fatigue pairs for these locations, before applying the bounding F_{en} of 4.524. The resulting CUF_{en} values were less than 1.0, thus eliminating these nickel alloy locations from further consideration. Nickel alloy welds in other reactor vessel locations were also similarly evaluated, and their CUF and CUF_{en} values were also less than 1.0 and lower values than the outlet and inlet nozzle to safe end weld results. Therefore the outlet and inlet nozzle to safe end weld results are considered representative and bounding.

   The Unit 1 RSG tubes location was also evaluated for the impact of the NUREG/CR-6909 curve, since applying the worst case difference in the curves could potentially increase the screening CUF_{en}'s to a value greater than 1.0. Therefore, the fatigue analysis of record was examined for this location. The stress analysis ranking evaluation revealed that the design basis CUF analysis of record for the Unit 1 RSG tube location was very conservative, since all transient cycles were assigned to a single maximum alternating stress. The fatigue analyses for other competing locations included at least several "lumped" or individual transients, with transient cycles distributed among several alternating stress ranges. This is a significantly less conservative method than the effect of assigning the less severe transient cycles (usually less severe transients have high cycles) to the most limiting alternating stress range, as was done in the Unit 1 RSG tubes evaluation. Because of this significant difference in fatigue methodology, a stress basis ranking bias was assigned, justifying that the tube location not be considered further in the screening. To preclude a more detailed comparison of the fatigue methodologies and further support the conclusion that the tubes location was not the most limiting RSG location, 60-year projected cycles were incorporated into the component fatigue analysis using the NUREG/CR-6909 curve to demonstrate a CUF_{en} less than 1. The resulting CUF using the NUREG/CR-6909 fatigue curve and the design transient cycles is 0.861 and the CUF_{en} is 3.90. Computing the CUF_{en} using the 60-year projected transient cycles results in CUF_{en} of 0.97. Since the resulting screening CUF_{en} was less than 1, the Unit 1 RSG tube component was eliminated from being a potential leading location in the EAF screening process.

The other four (4) eliminated locations, the Unit 2 OSG tubes, tube/tubesheet weld, and divider plate drain hole and fillet weld locations, have a lower CUF value as compared to the Unit 2 OSG primary chamber drain, which is also a nickel alloy location. All of these locations were evaluated with respect to the impact of the NUREG/CR-6909 fatigue curve for nickel alloys compared to the fatigue curve used in the CLB fatigue evaluations, in conjunction with the application of the nickel alloy screening F_{en}. The highest CUF_{en} remained at the Unit 2 OSG primary chamber drain, which was retained as a leading location with a CUF_{en} of 2.69.

3. Design basis CUF values are provided in the response to Request 1 above. The effect of the NUREG/CR-6909 fatigue curves on the CUF for the Unit 1 RSG tubes was included in conjunction with reducing conservatism and re-calculating CUF, to demonstrate the screening CUF_{en} was less than 1.0; and justify its elimination from further EAF consideration, as discussed in the response to Request 2 above. For the nickel alloy locations eliminated as indicated above, design basis CUF, revised CUF and CUF_{en} values accounting for the NUREG/CR-6909 fatigue curve and F_{en} of 4.524, are provided in the table below:

Component	Design Basis CUF	NUREG/CR-6909 (Revised) CUF	CUF_{en}
Units 1 and 2 Reactor Vessel Outlet Nozzle to Safe End Weld	0.11	0.13	0.59
Units 1 and 2 Reactor Vessel Inlet Nozzle to Safe End Weld	0.04	0.042	0.19
Unit 1 Replacement Steam Generator (RSG) Tubes	0.19	0.861	3.90 (0.97*)
Unit 2 Original Steam Generator (OSG) Tubes	0.233	0.23	1.02
Unit 2 OSG Tube to Tubesheet Weld	0.379	0.53	2.39
Unit 2 OSG Divider Plate (Drain Hole)	0.194	0.34	1.54
Unit 2 OSG Divider Plate (Fillet Weld)	0.193	0.36	1.61

*Using the 60-year projected transient cycles (See discussion in response to Request 2 above)

RAI 4.3.4-7, Derivation methods for determining maximum Fen as used in environmental fatigue calculations (060)

Applicability: Byron and Braidwood Stations

Background:

LRA Section 4.3.4 states that once the stress basis comparison has been performed and the leading transient section locations have been identified, screening environmental correction factors (F_{en}) are developed for each component so that cumulative usage factors including environmental fatigue, CUF_{en}, can be calculated.

Furthermore, the LRA states that those components with a screening CUF_{en} of less than 1.0 were removed from the list because they have been calculated using the design basis fatigue usage factors with a maximum F_{en} based on material.

Issue:

It is not clear whether the “maximum F_{en}” is the maximum calculated from the NUREG reports or whether is the maximum calculated for a particular transient section. The staff noted that if it is the latter, it is important to understand the applicant’s assumptions in calculating the maximum F_{en} based on material for a particular transient section.

Request:

Clarify if the maximum F_{en} based on the material is the calculated maximum F_{en} from the applicable NUREG reports or the calculated maximum from a particular transient section.

If the maximum F_{en} was based on the transient section, identify any assumptions (e.g., temperature, sulfur, dissolved oxygen, strain rate) used in calculating the F_{en} and the basis for these assumptions.

Exelon Response:

Two different maximum screening F_{en}s were calculated for each transient section. The first was calculated based only on material (material F_{en}). The second considered both the material and the maximum temperature of the transient section (temperature F_{en}). They were both calculated for a given transient section using the applicable NUREG reports.

To calculate the maximum Material F_{en}, bounding assumptions were made regarding the parameters (temperature, dissolved oxygen, sulfur content, strain rate) used in the F_{en} equations from the NUREG reports (NUREG/CR-6583, NUREG/CR-5704, and NUREG/CR-6909). Generally, parameters were chosen such that the maximum F_{en} penalty factor for the material would be calculated. The only exception is dissolved oxygen content. A value of 0.005 ppm was used for the dissolved oxygen (DO) content, which is typical of the PWR environment.

For PWR's, except for short periods during heatup/cooldown operations, the DO content is generally well below the 0.05 ppm criteria used to determine the transformed dissolved oxygen content parameter (O*) in the carbon and low alloy steel equations. A review of plant chemistry data from 2004 to 2012 supports the use of 0.005 ppm during normal operation and 0.05 ppm during heatup and cooldown operations. However, elevated DO content usually only occurs when reactor coolant temperature is low. During these periods of operations, fluid temperatures are in the range where the transformed metal temperature parameter (T*) = 0, and the applicable term in the applicable F_{en} equation exponent still reduces to zero. Therefore, in any case for PWR conditions, the term containing the O*T* product is zero in the F_{en} equations for carbon and low alloy steels. In the stainless and nickel alloy steel equations, use of PWR DO conditions is bounding.

The following assumptions were used to maximize the material F_{en} for the different materials:

<u>Austenitic Stainless Steels (NUREG/CR-5704):</u>

Max F_{en} = 15.348**** when:

Service Temperature, T = 200°C (392°F) or higher****

Dissolved Oxygen, DO = 0.005 ppm

Strain Rate, $\dot{\varepsilon}$ = 0.0004%/sec or lower

****below 392°F the Max F_{en} = 2.547

<u>Carbon and Low-Alloy Steels (LAS) (NUREG/CR-6583):</u>

Max F_{en} = 1.74 for Carbon and 2.455 for LAS when:

Sulfur Content, S = 0.015 weight percent or higher****

Dissolved Oxygen, DO = 0.005 ppm

Service Temperature, T = 350°C (662°F) or higher ****

Strain Rate, $\dot{\varepsilon}$ = 0.001%/sec or lower****

****since O* is 0.0 for PWR conditions S*, $\dot{\varepsilon}^*$, & T* are irrelevant for Carbon and LAS

<u>Ni-Cr-Fe- Alloys (NUREG/CR-6909):</u>

Max F_{en} = 4.524**** when:

Service Temperature, T = 325°C (617°F) or higher****

Dissolved Oxygen, DO = 0.005 ppm

Strain Rate, $\dot{\varepsilon}$ = 0.0004%/sec or lower

****Below 617°F T* is a function and max F_{en} must be calculated as a function of max temperature that the component experiences

The same assumptions were used to calculate the maximum temperature F_{en}, except that the maximum temperature for each transient section was input to the applicable NUREG equation. This still results in a bounding F_{en} for the material based on the temperatures in the transient section.

The maximum material F_{en} was used for the first step of the process to determine which components would have a CUF_{en} value less than 1.0 and could be eliminated from the screening process. For the next step in the process, the maximum temperature F_{en} for the transient section was used to determine if other components would have a CUF_{en} value less than 1.0, and could be eliminated from the screening process. The results of this effort were used in the later steps of the process to determine the governing component locations requiring more extensive EAF analysis. It should be noted that there were no equipment or piping systems in which every component location CUF_{en} was less than 1. Therefore, in all cases there was at least one location that was the most limiting. This means that it was not possible for a limiting equipment or piping location to be missed because it was prematurely eliminated.

RAI 4.3.7-1, Revisions to fatigue evaluations to account for thermal stratification (060)

Applicability: Byron and Braidwood

Background:

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," issued December 1988, requested utilities to demonstrate that the design requirements of the pressurizer surge line consider the effects for thermal stratification. LRA Section 4.3.7 states the demonstration was an ASME Section III fatigue analysis to account for thermal stratification that was identified as a TLAA.

The LRA further states that the original fatigue analyses for the pressurizer surge line included stratification sub-transients. The LRA states that the fatigue evaluations of the components affected by the bulletin were revised to consider transients in Table 4.3.1-1, 4.3.1 2, 4.3.1-4, and 4.3.1-5 and determined that the resulting cumulative fatigue usage will remain below 1.0.

Issue:

The applicant stated that the fatigue evaluations were revised to consider NRC Bulletin 88-11. However, it is unclear how the fatigue evaluations were revised to account for thermal stratification in applicable components.

Request:

1. Identify the pressurizer surge line stratification sub-transients in the CLB.

2. Identify which transients were considered when the fatigue evaluations were revised for components affected by Bulletin 88-11.

3. Confirm that the Fatigue Monitoring Program, when implemented, will monitor the redefined transient cycles and severities and will require action prior to exceeding design limits. If not, justify that the aging effects due to fatigue will be managed during the period of extended operation for the components impacted by these redefined transients.

Exelon Response:

1. The Byron and Braidwood current licensing basis surge line fatigue evaluation included pressurizer surge line stratification sub-transients, which were developed and identified in a plant specific evaluation in response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification". The plant specific evaluation was provided to the NRC in a letter dated January 8, 1991, Subject, "Zion Station Units 1 and 2 Byron Station Units 1 and 2 Braidwood Station Units 1 and 2 Supplemental Response to NRC Bulletin 88-11 Pressurizer Surge Line Thermal Stratification NRC Docket Nos. 50-295/304, 50-454/455, and 50-456/457" (Accession Number 9101140203). Tables 2-2a and 2-2b in this letter document that eleven (11) sub-transient cases were developed for the surge line piping and nine (9)

sub-transient cases were developed for the surge line nozzle. The sub-transients were developed based on a detailed evaluation, performed by the Westinghouse Owners Group (WOG), utilizing pressurizer surge line stratification data to characterize the cyclic activity during heatup and cooldown and defined a bounding set of differential temperatures. In the evaluation, the fatigue analysis for the pressurizer surge line was performed to account for the surge line pipe stratification sub-transients that occur during plant heatup and cooldown. These sub-transients were defined as a function of system ΔT (temperature difference between pressurizer water temperature and reactor coolant loop hot leg temperature) ranges, which were postulated to occur during 200 heatup and cooldown cycles. The sub-transients also included reactor coolant loop hot leg to surge line nozzle stratification at different ΔT's that result from reactor coolant pump trips postulated to occur during 200 heatup and cooldown cycles. Therefore, the sub-transients analyzed represent 200 plant heatup and cooldown cycles, distributed in defined ranges of system ΔTs.

2. In addition, the fatigue evaluations also included stratification effects in the applicable non-heatup/cooldown transients. No sub-transient cases were required for the non-heatup/cooldown transients. For the affected non-heatup/cooldown transients, stratification was assumed at the maximum applicable temperature difference for each transient cycle.

   The pressurizer surge line fatigue evaluation considered the transients documented in Tables 2-1, 2-2a, and 2-2b in the letter referenced in the response to Request 1 above. These transients are the same as the following transients in LRA Tables 4.3.1-1 and 4.3.1-4:

   Normal Conditions – transients 1 through 14;
   Upset Conditions – transients 20 through 30 and 34; and
   RCS Test Conditions – transients 37, 38, and 42.

3. The transients considered in the pressurizer surge line fatigue evaluation are documented in the response to Request 2 above. Only the heatup and cooldown transients in LRA Tables 4.3.1-1 and 4.3.1-4 (transients 1 and 2) were analyzed with refined sub-transients, in response to NRC Bulletin 88-11, as described in the response to Request 1. Since the existing heatup and cooldown transient cycle limits and definitions bound the refined analyzed sub-transients described in the response to Request 1, no transients monitored by Fatigue Monitoring (B.3.1.1) aging management program were redefined. The transient limits and associated definition of the transients documented in the response to Request 2 above will be monitored by Fatigue Monitoring (B.3.1.1) aging management program implementing procedures. Therefore, the Fatigue Monitoring (B.3.1.1) aging management program will monitor the transient cycles and severities and will require action prior to exceeding design limits to account for thermal stratification in the pressurizer surge line.

Enclosure C

Application for Withholding Proprietary Information from Public Disclosure

Supporting the following Set 10 RAI Responses

RAI 4.3.1-1
RAI 4.3.1-3
RAI 4.3.4-1
RAI 4.3.4-2

Notes:

1. The Proprietary versions of the responses are contained in Enclosure A.

2. This Enclosure consists of this cover page and seven pages associated with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure.

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CAW-14-4026

September 9, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-PAFM-14-31, Revision 3, Attachment 1, "Byron and Braidwood Units 1 and 2 License Renewal: NRC Request for Additional Information Responses," (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-14-4026 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Exelon.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference CAW-14-4026, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

JA Gresham

James A. Gresham, Manager

Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 9th day of September 2014

Anne M. Stegman

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Anne M. Stegman, Notary Public
North Huntingdon Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-PAFM-14-31, Revision 3, Attachment 1, "Byron and Braidwood Units 1 and 2 License Renewal: NRC Request for Additional Information Responses," (Proprietary), for submittal to the Commission, being transmitted by Exelon letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the NRC letter, "United States Nuclear Regulatory Commission Letter ML14038A336, 'Requests For Additional Information for the Review of the Bryon Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2, License Renewal Application – Aging Management – Set 10 (TAC Nos. MF1879, MF1880, MF1880, MF1881, and MF1882'" and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Perform Environmental Fatigue Screening

(ii) Utilize the Westinghouse Reference Fatigue Database

(iii) Utilize Plant Operating Data in lieu of Design Transient Data

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of the information to its customers for the purpose of performing required environmental fatigue screening and fatigue evaluations, and utilizing plant operating data in lieu of design transient data.

(ii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar environmental fatigue-related evaluations and plant operating data utilization without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC that associated with the NRC letter, "United States Nuclear Regulatory Commission Letter ML14038A336, 'Requests For Additional Information for the Review of the Bryon Nuclear Station, Units 1 and 2, and Braidwood Nuclear Station, Units 1 and 2, License Renewal Application – Aging Management – Set 10 (TAC Nos. MF1879, MF1880, MF1880, MF1881, and MF1882'" and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).