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RS-17-119 10 CFR 2.206

September 1, 2017

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

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

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

Subject: Response to Request for Voluntary Response to Petition Regarding Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Units 1 and 2 High Energy Line Break in Main Steam Isolation Valve Room

Reference: Letter from Mr. Joel S. Wiebe (U.S. NRC) to Mr. Bryan C. Hanson (Exelon Generation Company, LLC), dated July 26, 2017

Pursuant to your request, Exelon Generation Company, LLC (EGC) is providing a voluntary response to the request for information related to the matter described in the referenced letter. The referenced letter requested a written response to be submitted within 30 days. Accordingly, the response is due August 25, 2017. However, based on discussion between EGC and NRC, it was agreed that the response would be submitted on or before September 1, 2017. A detailed response to each of the specific concerns contained in the request is provided in the attachments to this letter.

Due to the nature of the requests for items a-i, EGC has provided the responses in two attachments. The responses for items a-g and i are technical in nature and have responses provided in Attachment 1. The responses for item h have been generated by independent evaluators using interviews and available documentation, with the discussion and responses provided in Attachment 2.

The responses for Attachment 1 contain numerous references, which have been provided as an Enclosure. The enclosure contains Corrective Action Program (CAP) Issue Reports (IRs), Operability Evaluations, and two calculations. The IRs and Operability Evaluations were prepared for internal EGC use only, and therefore the names of individuals identified in the IRs have been redacted as Personally Identifiable Information (PII) under 10 CFR 2.390, "Public

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inspections, exemptions, and requests for withholding." The purpose of CAP is to provide all EGC nuclear employees with a means by which to freely and openly identify issues or concerns. Once identified in an IR, the issue or concern is evaluated, appropriate corrective actions are identified, and the issue is tracked until resolved or all actions are completed. EGC requests that the names of the individuals identified in the IRs be withheld to protect the personal privacy of those individuals and the integrity of its CAP and self-evaluation process.

There are no regulatory commitments contained in this letter. Should you have any questions regarding this letter, please contact Ryan Sprengel at (630) 657-2814.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachment 1: Response to Request for Voluntary Response, Items a-g and i
Attachment 2: Response to Request for Voluntary Response, Item h

Enclosure: Reference documents for Response to Request for Voluntary Response (CD)

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, Braidwood Station (w/o Enclosure)
NRC Senior Resident Inspector, Byron Station (w/o Enclosure)

Response

- Item a: Provide a copy of Revision 3 to calculation 3C8-0282-001, "Main Steam Tunnel Pressure Study for Main Steam Line Break [MSLB)," and any subsequent revisions. Also provide any documents or calculations that are used as input to the calculation and are necessary to understand the calculation. The petitioner states that the piping volumes obtained from the plant layout between the affected steam generator and the various steam line isolation valves, feedwater isolation valves, and control valves given in UFSAR Section 6.2.1.1.3 were appropriately included in the analysis for the main steam line break inside containment; and were not included in the analysis of record for the breaks outside the containment. Provide the calculations that document the main steam piping volume input and assumptions used in the analysis for breaks inside the containment and for breaks outside the containment. In case the volume calculation for the break inside the containment is included in the main steam line break containment pressure response analysis, provide the calculation. Use of an electronic system that allows the NRC staff to view these documents is preferred in lieu of transmitting large amounts of paper.

  Response:

  The following documents are provided as an enclosure to this response:
  1. 3C8-0282-001 Revision 3
  2. Framatome 32-1239267-01

  Design Analysis 3C8-0282-001 is the design basis analysis for the structural design of the Main Steam Isolation Valve (MSIV) house and the Main Steam (MS) Tunnel. The analysis determines the maximum pressure in the affected areas following a Main Steam Line Break (MSLB) outside containment. The results of the analysis are compared to the design basis structural analysis that qualifies the structures for bounding pressures.

  The analysis models a double ended rupture of the Main Steam Line at different locations inside the MSIV room and in the MS Tunnel.

  Volume used for MS piping that cannot be isolated:

  MSLB Inside Containment

  Braidwood U-1 MSLB Inside Containment – CN-CRA-13-29 Rev. 0 (BRW UHS Higher Temperature)
  Maximum Unisolable MS volume (MSIV failure case) - 11,576 ft^3
  Maximum Unisolable Feedwater (FW) Volume (FW isolation valve (FWIV) failure case) – 701.7 ft^3

  Braidwood U-2 MSLB Inside Containment – CN-CRA-13-21 Rev. 0 (BRW UHS Higher Temperature)
  Maximum Unisolable MS volume (MSIV failure case) - 11,576 ft^3
  Maximum Unisolable FW Volume (FWIV failure case) – 654.9 ft^3

Byron U-1/U-2 MSLB Inside Containment – CN-CRA-10-29 Rev. 0
Maximum Unisolable MS volume (MSIV and FWIV failure case) - 11,576 ft^3
The actual unisolable steam line volume for a FWIV failure is 767 ft^3; the higher value is conservative.

CN-CRA-12-29, CN-CRA-13-21, and CN-CRA-10-29 are proprietary and not provided as a part of this response.

MSLB Outside Containment

There are two analyses done for MSLB outside containment. One analysis maximizes the calculated temperature and determines the Environmental Qualification (EQ) temperatures for the equipment in the Main Steam Safety Valve (MSSV) room (MSIVs, Power Operated Relief Valves, FWIVs). This analysis is done by Westinghouse. The other analysis, 3C8-0282-001, Rev. 3, determines the maximum pressure in the affected areas following a MSLB outside containment.

EQ Temperature Analysis
Design Analysis CN-CRA-10-29 Revision 0 (Byron U-1 and U-2)
Design Analysis CN-CRA-10-29 Revision 1 (Braidwood U-1 and U-2)

EQ Temperature Analysis – Auxiliary Feedwater Delay
Design Analysis LTR-PL-13-32 Revision 1 (Byron U-1 and U-2)
Design Analysis CN-CRA-13-5 Revision 1 (Braidwood U-1 UHS Higher Temperature)
Design Analysis CN-CRA-13-6 Revision 1 (Braidwood U-2 UHS Higher Temperature)

CN-CRA-10-29, CN-CRA-13-5, CN-CRA-13-6, and LTR-PL-13-32 are proprietary and not provided as part of this response.

Structural Design Pressure Analysis

Design Analysis 3C8-0282-001 Revision 3 uses Mass and Energy (M&E) from:
Unit 2 – Byron / Braidwood Preliminary Safety Analysis Report (PSAR) Table 6.2-15a – This is the M&E for the MSLB inside containment.
Unit 1 – Appendix B Framatome Report 32-1258100-00 dated 8/15/96 (Attached to NFS TODI 960136 Rev. 0, which summarized results of 32-1239267-00)

Design Analysis 3C8-0282-001 Revision 3 is the design basis analysis for the structural design of the MSIV house and the MS Tunnel. The analysis determines the maximum pressure in the affected areas following a MSLB outside containment. The Unit 1 mass and energy release is taken from the MSLB inside containment analysis. This analysis does include the volume of the steam lines (Analysis 32-1258100-0, which references 32-1239267-00).

The Unit 2 mass and energy release is taken from PSAR Table 6.2-15a. This is the energy input for the analysis of the MSLB inside containment.

Design Analysis 3C8-0282-001 Revision 4 has been prepared/reviewed by Sargent and Lundy (S&L), but not yet accepted by Exelon. As described later, actions are in progress to finalize this calculation as the analysis of record.

- Item b: Provide a copy of Issue Report 792213, "MSLB Calc[ulation] [sic] Energy Release Error," dated June 30, 2008. Also provide any documents that document action taken in response to this Issue Report or provide closure of any actions. Provide any follow up or related Issue Reports or documents that document action taken or provide closure of related Issue Reports. Draft or uncompleted documents need not be provided. Provide a discussion or description of the timeline, if necessary to aid in the understanding of the sequence or extent of actions taken in response to this issue or related issues associated with this Issue Report.

  Response:

  The MSIV room floor had a ¼ inch diamond plate hatch with a rubber seal that protects the Auxiliary Feedwater (AF) Tunnel from flooding. The floor plate was held up by angle iron secured with concrete anchors to the hatch vertical walls below. The original structural basis of the plate was to seal the AF Tunnel from a flooding event in the MSIV room and tunnel. However, it was discovered that AF Tunnel hatch had never been evaluated against the MSLB design basis MSIV room pressure. The calculation 3C8-0282-001, Revision 3 design basis pressure profile was higher and increased faster than the flood pressure. While performing the operability evaluation on the original AF Tunnel flood plate, an error in the Unit 1 mass and energy table was discovered in the 3C8-0282-001, Revision 3 calculation. Additionally, other issues were discovered in the 3C8-0282-001, Revision 3 inputs that need to be corrected. The AF Tunnel hatches were all strengthened to a new bounding pressure load profile. Upon identification of these errors, Issue Reports (IR) and associated actions were written to correct calculation 3C8-0282-001, Revision 3. These actions are still active and underway.

  (Note: The corollary Braidwood IR is 792215)

  See following Table for event timeline. All referenced IRs and Operability Evaluations are provided via Enclosure.

Date	Event Discussion
7/24/2007 (BYR) 7/26/2007 (BRW)	Braidwood (BRW) IR 654270 – Operability Evaluation 07-007 Byron (BYR) IR 653093 – Operability Evaluation 07-006 (EC 366685) The design analysis of the AF Tunnel flood seal hatches did not include loads due a High Energy Line Break (HELB) on qualification of the

	hatches and supporting structure Concrete Expansion Anchors (CEAs). The AF Tunnel hatches are required to remain intact and perform a sealing function between the AF Tunnel and the MSIV rooms.
7/27/2007 (BYR) 7/31/2007 (BRW)	Braidwood (BRW) Operability Evaluation 07-007 (ATI 654270-02) Byron (BYR) Operability Evaluation 07-006 (EC 366685 Rev. 0) The operability evaluation uses the results of HELB analysis 3C8-0282-001 Revision 3. Operability is supported with a low factor of safety for the installed CEAs on the AF Tunnel access opening cover plate. Corrective Action – Install a design change to restore full design margins for the AF Tunnel access opening cover plate CEAs.
6/23/2008 (BYR) 6/24/2008 (BRW)	IR 789344 (BYR) IR 789791 (BRW) Errors are found with the MS Tunnel pressurization analysis (3C8-0282-001 Revision 3) with respect to credited vent paths. Actions are established to determine maximum tunnel pressures without vent paths.
6/30/2008	IR 792213 (Byron) IR 792215 (Braidwood) EC 371293 is completed to develop best estimate M&E release rates. In the processing of EC 371293, an error was discovered in the Unit 1 MSLB outside containment energy release calculation. The erroneous energy release inputs were used in calculation 3C8-0282-001 Revision 3 for Braidwood Unit 1 and Byron Unit 1. The energy release inputs are thermodynamic internal energy U, not the required input enthalpy. For M&E to be correctly calculated, enthalpy has to be used, otherwise it is missing the kinetic energy of the break flow stream. Enthalpy is defined as U+PV. The analysis used the raw internal energy data since the post-processed enthalpy data was not listed in the calculation. The 3C8-0282-001 Revision 3 calculation main body, which was now applicable to Unit 2, had the original Westinghouse M&E which was bounding. EC 371293 developed best estimate short term (0-1.5 seconds) M&E releases for a MSLB outside containment in the MSIV room for the Byron and Braidwood plants for the purpose of evaluating compartment pressurization. EC 371293 calculated correct energy release values for Units 1 and 2 using enthalpy values.
7/1/2008	S&L Evaluation 2008-11287 analyzed pressurization effects of a break in the Lower Safety Valve Room, using the M&E from EC 371293. This analysis does not include previously credited vent paths via blank-off plates in the MSIV room (Reference timeline entries for 6/23/2008 and

	6/24/2008). The analysis determined maximum pressures of 21.7 psig for Braidwood and Byron Unit 1 and 19.4 psig for Braidwood and Byron Unit 2.
7/2/2008	EC 366723, Revision 2 (BYR)/EC 371345, Revision 1 (BRW) evaluates AF Tunnel Flood Barrier for maximum HELB loads (21.7 psi for Unit 1 and 19.4 psi for Unit 2).
7/2/2008 (BRW)	Operability Evaluation 07-006 (BYR) and 07-007 (BRW) were revised to include the maximum pressures from S&L Evaluation 2008-11287. Documented in: ATI 789791-03 (BRW), EC 366685 Rev. 3 (BYR)
4/14/2008 (BRW-2) 4/23/2008 (BRW-1)	The design for the modification to AF Flood Hatch cover is qualified for a 40 psi HELB Load, 20 psi HELB pressure with a Dynamic Load Factor, DLF, of 2. The actual DLF can be calculated with a possible maximum value of 2. The DLF for the evaluation in EC 366723 (BYR)/EC 371345 (BRW) was calculated to be below 1.1. Therefore, the AF Flood Hatch cover qualification analysis is conservative because it uses the maximum possible DLF. The total qualification HELB pressure of 40 psi bounds the maximum calculated pressure of 21.7 psi from S&L Evaluation 2008-11287 with a DLF of < 1.1. (BRW) Design Analysis 5.6.3-BRW-08-0045-S (Unit 1), 5.6.3-BRW-08-0040-S (Unit 2).
8/7/2008 (BRW)	ATI 792215-03 initiated to track completion of design analyses for mass and energy release and Steam Tunnel Pressurization.
12/20/2008 (BRW-1)	Unit 1 Modified AF Flood Seal Cover Supports installed (EC 369245)
12/18/2008 (BRW-2)	Unit 2 Modified AF Flood Seal Cover Supports installed (EC 369246).
3/6/2009 (BYR)	At Byron Station, the design for the modification to AF Flood Hatch cover is qualified for a 50 psi HELB Load (25 psi HELB pressure with a conservative Dynamic Load factor of 2). Design Analysis 5.6.3.9-BYR08-065
3/12/2009 (BRW-1 and -2)	Operability Evaluation 07-007 is closed at Braidwood Station.
3/13/2009 (BYR-1)	Modified AF Flood Seal Cover Supports installed (EC 367209, EC 367210).
3/26/2009 (BYR-2)	Modified AF Flood Seal Cover Supports (MSIV Room 2A and 2D) installed (EC 367211).
3/27/2009 (BYR-2)	Modified AF Flood Seal Cover Supports (MSIV Room 2B and 2C) installed (EC 367212).
7/1/2009 (BYR-1 and -2)	Operability Evaluation 07-006 is closed at Byron Station.

2010-2013	ATI 792215-03 for Braidwood Station; ATIs 653093-19, 1284054-03, and 1531420-02 for Byron Station Actions remained to track the revision to the MS tunnel pressurization analysis (3C8-0282-001 Revision 3). Significant conservatism is used to qualify the modified AF Tunnel Hatch covers, not expected to be affected by the revised MS Tunnel pressurization analysis.
2/27/2013	Contract is approved for revising analysis 3C8-0282-001 Revision 3. Sargent & Lundy is assigned the work.
9/13/2013	A new mass and energy release calculation FAI/12-0550, Revision 1 (Fauske analysis) is completed, "Byron and Braidwood Turbine Building Refined HELB Mass and Energy releases."
11/12/2013	Calculation 3C8-0282-001 Revision 4 was prepared and reviewed by S&L. Calculation was sent to Exelon for review. This calculation used mass and energy release inputs from the Fauske analysis of a double ended guillotine pipe break. Calculation 3C8-0282-001 Revision 4 has not been approved by the vendor and has not been accepted by Exelon. The calculation 3C8-0282-001 Revision 4 credits venting through the opening in the roof of the MSIV room after the concrete slab lifts. The credited vent area is conservative. The roof of the MSIV house at Byron has three (3) concrete slabs for each of the two (2) openings; the roof of the MSIV house at Braidwood has one concrete slab for each the two (2) openings. The analysis credits venting through both openings due to the lift of three smaller concrete slabs at Byron, only one third of the available area. The detailed response of the slab lifting is not analyzed.
2014-2015	Exelon was reviewing the design and licensing basis for the break size and its location for the MSLB outside containment event. Enercon was contracted to perform an analysis for a crack equivalent to the flow area of a single ended pipe rupture directly outside the MSIV house. The intent was to analyze a smaller break size, similar to that of a single ended break that is part of the Byron and Braidwood Licensing Basis. FSAR Q&A 10.4 requested to evaluate a crack equivalent to a single ended pipe rupture.
12/15/2015	Analysis BRW-15-104-M/BYR15-11 is completed by Enercon. The analysis is a new MS Valve Room Pressurization Analysis based on the single ended pipe rupture.
2016	Analysis BRW-15-104-M/BYR15-11 is reviewed by Exelon (Site and Corporate).
2016	Reviews delayed by personnel changes (retirement and transfers).
9/20/2016	Pre-job brief held between Byron, Braidwood, and Corporate Engineering for completion of Owner Acceptance Review of analyses 3C8-0282-001 Revision 4 and BRW-15-104-M/BYR15-111.
11/8/2016	Weekly meeting scheduled between Byron, Braidwood, and Corporate Engineering to track issues and progress on Owner Acceptance Review.
2/4/2017	2.206 Petition letter issued.
Present	Exelon plans to complete calculation 3C8-0282-001 Rev. 4 for maximum pressure analysis.

Item c: The petitioner questioned using an incorrect main steam line break area of 1.4 ft^2 instead of 5.6 ft^2; and an incorrect main steam isolation valve closure time. Provide a response to the petitioner's concern documented in his following email to engineering management dated January 27, 2017:

> " .... Section C3.6 of the UFSAR analyzes double-ended break in the MSIV [main steam isolation valve] room; one could posit that our CLB [current licensing basis] is overly conservative but its presence cannot be denied. The table in section C3.6 lists the break size as 1.4 ft^2, which may seem supportive of question 010.04 [UFSAR Question 010.04]. However, there are nuances to this value that are revealed upon examination of the flowrates as a function of time; the unit 2 table is used since the unit 1 has errors. At time zero, the total flow is 11,000 lbm/sec. The flow is saturated (initially) and choked. A determination of Moody choked flow requires a pressure and enthalpy. Since the flow is saturated the enthalpy can be used to find the pressure, doing so gives a value of 948 psia. Using the Moody critical flow tables, this corresponds to 1963 lbm/ft^2-sec. Dividing the initial flow of 11000 lbm/sec by 1963 gives a break area of 5.6 ft^2, obviously not the 1.4 ft^2 listed in the table, but it does correspond to (4 SG * 1.4 ft^2 flow limiter). Another salient point in the table is what happens between 10.0 and 10.1 seconds; at 10 seconds the value is 9318 lbm/sec and 0.1 seconds later is has dropped substantially to 2098 lbm/sec. The latter flow value is 22.5% of the former and is what would be expected when the MSIVs (each feeding through a 1.4 ft^2 restrictor) on the 3 non-faulted generators are isolated by their MSIVs.
>
> Based on the above, I [the petitioner] infer that the CLB break is double-ended in the MSIV room and fed by 4 SGs until MS IVs close at 10 seconds. Since the peak pressures occur with the first second, use of 1.4 ft^2 from time zero is not consistent with the CLB."

Response:

IRs 4046781 and 4046785 identify the need to correct the 1.4 ft^2 area in UFSAR Section C3.6. However, the M&E tables used as input to the 3C8-0282-001, Rev. 3 analysis of record were for a double ended rupture as the petitioner concluded.

The licensing basis for the Byron and Braidwood plants for a MSLB outside containment is the following:

1. Inside the MSIV rooms - A crack equivalent to the flow area of a single ended pipe rupture (Reference FSAR Q&A 10.4)
2. Outside the MSIV rooms – Double Ended Guillotine Break

For the purpose of calculating maximum pressures within the MSIV room, Exelon plans to complete design analysis 3C8-0282-001 Revision 4. This current draft analysis uses M&E values from Fauske analysis FAI/12-0550 Revision 1. The current draft M&E analysis uses a double ended rupture downstream of the MSIVs. This is bounding for the required break size inside the room, i.e., a crack equivalent to the flow area of a single ended pipe rupture (Reference FSAR Q&A 10.4). The piping diameter at the

break location is 32.75 inches and the flow area is 4.9 ft^2 on each side of the break for a total flow area of 9.8 ft^2. The M&E release transient was analyzed for both SG flow limiter nozzle sizes, 1.1 ft^2 for Unit 1 and 1.4 ft^2 for Unit 2. Therefore, the M&E release is limited by the size of the flow limiting nozzles. The four SG limiting flow area is 4.4 ft^2 for Unit 1 and 5.6 ft^2 for Unit 2. The above described approach continues to be under evaluation by Exelon and final analyses will be available for inspection.

With regard to the MSIV isolation time, calculation 3C8-0282-001, Rev. 3 analyzed only the peak pressurization period. The peak pressurization period analyzed for Unit 1 was the first 3.0 seconds, and for Unit 2 it was the first 1.0 seconds. Both Unit 1 and Unit 2 M&E tables do not credit MSIV isolation during the peak pressure period analyzed and are conservative in that regard. The isolation of the 3 intact SGs at 10.1 seconds mentioned by the petitioner has no impact on the results because the peak pressure occurs in the first second.

The pressure transient analysis in calculation 3C8-0282-001 Revision 4 locates the break inside the MSIV room. All calculated pressures are within the design pressures used in the design basis structural analysis.

Revision 4 to design analysis 3C8-0282-001 has been prepared and reviewed by a contractor (S&L) and is under review by Exelon. The analysis credits venting through the roof opening as the slab lifts during the MSLB event. The calculation is conservative as it only credits the opening area of the smallest individual concrete slab from Byron Station (i.e., 20.5 ft^2 in each opening, for a total of 41ft^2 crediting the two openings in the MSIV room nearest the break). The opening area for the two openings in each MSSV room is over 120 ft^2 at both Byron and Braidwood.

- Item d: Provide any operability evaluations performed in response to any errors found in the calculation or taken in response to conditions found while taking actions to resolve the errors. If none were performed provide an explanation.

  Response:

  Operability Evaluation 07-006 for Byron (EC 366685 Rev. 3) and 07-007 Rev. 4 for Braidwood (ATI 789791-03).

- Item e: Provide a description of actions that are currently in progress to resolve any errors identified in the calculation, including the scope of the actions and target dates for completing the actions. Provide a description as to how these actions are being tracked to completion. If Issue Reports are involved in the tracking, provide the link to those identified in your response to "b," above or provide a copy, along with the supporting documents if they were not included in "b," above.

- Item f: Provide a description of the methodology being used for resolution of any errors found in the calculation. Provide the basis for its use and if and when the NRC has approved such methodology for this application.

  Response:

  See below for detailed responses to Items (e) and (f).

  Error in Calculation: Design Analysis 3C8-0282-001 Revision 3 uses incorrect enthalpy for Mass and Energy Release
  CAP Issue Reports: 792213 (Byron), 792215 (Braidwood)
  Action to Resolve: Revision 4 to design analysis 3C8-0282-001
  Action Tracking Assignments: 792215-03 (Braidwood) targeted for 9/29/2017, 1531420-02 (Byron) targeted for 12/29/2017
  Methodology Used for Resolution: Revision 4 of 3C8-0282-001 uses 3 seconds of M&E values from Fauske Analysis FAI/12-0550. It is a new RELAP5, MOD3.3 model of the steam generators and included all the main steam piping volumes, both active and inactive available to supply steam to the break location. RELAP5, MOD3.3 is commonly used for developing M&E for compartment analysis and has been approved by NRC for many similar applications. The current methodology for M&E inputs to the 3C8-0282-001 calculation was not identified by NRC in licensing correspondence as essential as a basis for approval that would need a license amendment to change. The current M&E methodology basis for Unit 1 is Framatome RELAP5, MOD2, and for Unit 2 is from the PSAR MSLB inside containment. Changing from these to a new RELAP5, MOD3.3 is not an element of the methodology that would require NRC review and approval. The RELAP5 version being used has corrected the Framatome error by using an appropriate total flowing enthalpy rate from the break, which is divided by the total mass flow rate to obtain the appropriate enthalpy value for each time step.

  Error in Calculation: Design Analysis 3C8-0282-001 Revision 3 uses vent area for blow-out plates that were found to be blank-off plates
  CAP Issue Reports: 789344, 790428 (Byron); 789791 (Braidwood)
  Action to Resolve: Revision 4 to design analysis 3C8-0282-001
  Action Tracking Assignments: 792215-03 (Braidwood) targeted for 9/29/2017, 1531420-02 (Byron) targeted for 12/29/2017
  Methodology Used for Resolution: Revision 4 of 3C8-0282-001 uses the as-built configuration from plant walkdowns at both Byron and Braidwood to correct the error. The analysis uses a model that is bounding for Byron and Braidwood. The compartment pressure calculation uses the same RELAP4, MOD5 computer code methodology that was identified in the UFSAR.

- Item g: By letter dated August 2, 2012 (ADAMS Accession No. ML12208A338), the NRC requested in item e.1. the plan and schedule for the extent of condition review of high-energy line break areas other than the Turbine Building. Explain why the issue with calculation 3C8-0282-001 was not resolved based on the extent of condition review.

  Response:

  The EGC response to the request for additional information from August 2, 2012 (ADAMS Accession No. ML12208A338), was provided in letter RS-12-145 dated August 31, 2012. In the response, EGC only identified that the Measurement Uncertainty Recapture (MUR) project had not identified any issues other than the issues with the Turbine Building HELB. The response identified that there were continuing actions related to the Turbine Building HELB and new issues would be addressed as identified.

  By letter dated December 6, 2012 (ADAMS accession No. ML12271A308) additional request for information was made regarding resolution of the Turbine Building HELB non-conformance, including details of the Extent of Condition reviews performed. EGC provide the response to this RAI in letter RS-13-189 dated July 5, 2013. In response to information requested under item 2.c, the following was identified:

  > EGC has concluded that the Auxiliary Building, MSIV Rooms/MS Tunnels, and Containment Building HELB AORs are consistent with the current licensing basis and were performed in accordance with approved methodologies, and do not contain the non-conformances identified in the TB HELB analyses. Therefore, the environmental qualification parameters previously evaluated have not changed and the conclusions of the MUR Power Uprate submittal(s) (References A-1, A-2, A-3, and A-4) related to the equipment environmental qualifications resulting from a postulated HELB continue to be valid.
>
> Note, during the extent-of-condition review, some gaps or document deficiencies were identified. These gaps and deficiencies, summarized below, have been entered into the Braidwood and Byron Corrective Action Program for evaluation and disposition. These deficiencies do not impact the extent-of-condition conclusions or the conclusions of the MUR analyses related to equipment environmental qualifications.
>
> ...
>
> - Design analysis for MSIV Room (also referred to as the MS Safety Valve Room) pressurization is being revised to apply a refined mass and energy release, more accurately reflect the configuration of the MSIV rooms and vent paths, and to use state of the art software. This update is in progress to address a condition identified prior to the HELB issue. Actions are in place to track completion (Byron IR 1531420, Braidwood IR 792215).

- Item i: In addition to the information requested above, if the technical issues in item 4 of the Petition Detailed Discussion are not addressed in items a through f, above, provide a discussion of the issue resolution, including scope and target dates if resolution is ongoing.

  Response:

  All technical issues identified in item 4 of the Petition Detailed Discussion have been addressed in the responses to items a through f above.

I. Introduction

By letter dated July 26, 2017, the NRC requested Exelon Generation Company, LLC (EGC) to evaluate the following issues:

- *Item h: Provide your evaluation of the following information from the February 8, 2017, Petition.*

  Detail 1: Exelon management stated that information in the updated final safety analysis report (UFSAR), Section C3.6, being used by the petitioner to support a technical position was excessive detail and could be removed in accordance with Nuclear Energy Institute guidance. When the petitioner stated he did not think it appropriate to remove the information, the Byron manager directed Braidwood personnel to remove the information.

  Detail 2: During a conference call on January 31, 2017, the petitioner pointed out that the UFSAR, Section 3.6.2.1.2.1.2 is inconsistent with another section of the UFSAR in the use of the Break Exclusion Zone concept. Without discussion or review of the evidence supporting the position, the Byron manager dismissed the internal inconsistency by saying that the information supporting the position could be deleted as an UFSAR cleanup item.

  Detail 3: Less than a month before the meeting discussed in Detail 2, there was an operability concern where Engineering management maintained a position of operability in the face of conflicting information.

  Detail 4: Details 1 through 3, above, support a conclusion that Exelon management cherry picks information to support operability and dismisses contrary views.

  The documented results of your evaluation should include sufficient information for the NRC to determine: (a) if the concern was substantiated; (b) that the organization or individual conducting the evaluation was independent of the concern and was proficient in the related functional area; (c) that the evaluation was of sufficient depth and scope to determine that the appropriate root causes and generic implications were considered; (d) that the corrective actions, both planned and completed, were sufficient to correct the specific example and generic implications and to prevent recurrence; (e) if your evaluation identified any compliance issues with NRC regulatory requirements or commitments, the corrective actions taken or planned, and the corrective action document that addressed the issues; (f) if interviews of individuals were conducted as part of your review, the basis for determining that the number and cross section of individuals interviewed, as well as the scope of the interview, was appropriate to obtain the information necessary to fully evaluate the subject concern, and the interview questions used; and (g) if your evaluation included a sample review of related documentation and/or potentially affected structures, systems, and components, your response should include the basis for determining that the selected sample size was appropriately representative and adequate to obtain the information necessary to fully evaluate the concerns. The NRC will consider these factors in reviewing the adequacy of your evaluation.

II. Response

A. Evaluators

The evaluation for this response was performed in multiple parts. The lead evaluator, and primary evaluator for Detail 3, is a Senior Regulatory Engineer in the EGC Corporate Licensing department. The evaluator has 15 years of nuclear experience in training, engineering, and licensing. The evaluator is independent of the Engineering Department, nuclear plants' organizations, and the events under evaluation.

The evaluation for Details 1 and 2 was primarily performed by a Principal Regulatory Engineer at an EGC Nuclear Power Plant. The evaluator has 32 years of nuclear experience in engineering and licensing. The evaluator is independent of the Engineering Department and the events under evaluation.

The evaluation for Detail 4 was primarily performed by an Employee Concerns Program (ECP) Representative assigned to Byron Station, and is independent from the Engineering organization. Duties include providing governance and oversight of ECP for the Byron Station, and conducting investigations of concerns that fall within the scope of the ECP. The evaluator has approximately 3 years of experience conducting ECP investigations and over 14 years of experience in the nuclear industry.

B. Methodology

The primary evaluator gathered and reviewed relevant documentation, including Corrective Action Program Issue Reports (IRs), Engineering Changes (ECs), licensing basis information, procedures, and ECP investigation results. The primary evaluator also discussed this matter with the additional evaluators and shared the relevant documentation.

For Detail 4, the ECP Representative performed interviews with seven individuals who were involved in the meetings and topics identified in Details 3 and 4. These interviews included personnel known to have been party to the meeting on January 31, 2017 and a part of discussions related to the operability concern identified in Detail 3. Interviews were conducted using a standardized set of questions.

C. Background

Interview questions used by the ECP Representative are included below:

Questions related to details 1 and 2:

1. Do you recall a conference call that was held on 1/31/17 where a double ended break in an MSIV room was discussed?
2. Tell me what you recall about that call?
3. How did the manager respond to a concern raised on the double ended break in an MSIV?
4. How was the response perceived?
5. Was this response viewed as opposition to the person with the concern?
6. Who was the Manager?
7. Did the Manager direct Braidwood Engineering to update the USFAR?

8. Tell me what you recall of the discussion of the Break Exclusion Zone Concept from the call?
9. What was the Managers response?
10. How was that response perceived?
11. Was this response viewed as opposition to the person with the concern?
12. How does management react to dissenting views when operability is a concern?
13. How are these items addressed?
14. Do you feel that you can raise nuclear safety / nuclear quality concerns without fear of retaliation?
15. Is there anything else you would like to add?
16. Based on the questions I have asked is there any other information that you believe may be pertinent in this investigation?

Questions related to detail 3:

1. Do you recall a meeting about the seismic qualified relays back in January of this year?
2. Tell me what you recall about that meeting?
3. How did the manager respond to this concern?
4. How was the response perceived?
5. Was this response viewed as opposition to the person with the concern?
6. Who was the Manager?
7. Was the concern dismissed by management?
8. What was your impression of the meeting?
9. Was the concern resolved?
10. How does management react to dissenting views when operability is a concern?
11. How are these items addressed?
12. Do you feel that you can raise nuclear safety / nuclear quality concerns without fear of retaliation?
13. Is there anything else you would like to add?
14. Based on the questions I have asked is there any other information that you believe may be pertinent in this investigation?

D. Evaluation

Detail 1:

Exelon management stated that information in the updated final safety analysis report (UFSAR), Section C3.6, being used by the petitioner to support a technical position was excessive detail and could be removed in accordance with Nuclear Energy Institute guidance. When the petitioner stated he did not think it appropriate to remove the information, the Byron manager directed Braidwood personnel to remove the information.

Assessment:

UFSAR Section C3.6 was provided in response to FSAR NRC Question 010.4. Responses to NRC Questions were incorporated during the conversion from the FSAR to the UFSAR. UFSAR Section C3.6 was last updated in UFSAR Revision 7 associated with the Unit 1 Steam Generator Replacement Project and included appropriate regulatory review documentation (i.e., various 50.59 Reviews for different aspects of the change).

Appendix A of NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1 provides guidance for making voluntary modifications to updated FSARs (UFSARs) (i.e., removal, reformatting and simplification of information, as appropriate) to improve their focus, clarity and maintainability.

NEI 98-03, Section A2 states the following:

> *It is the intent of this guideline to help licensees remove unimportant information from UFSARs such as excessive detail, obsolete information, or redundant information. This guideline is not intended to be used to remove important information from UFSARs about features or functions of SSCs that insights from operating experience or probabilistic risk assessments indicate are risk-significant. The intent that risk-significant information be retained does not preclude removal of obsolete or redundant information, or excessive detail concerning the design or operation of risk-significant SSCs, provided that the action is consistent with the guidance in this Appendix.*

The following additional guidance is provided Section A4.1, "Removing Excessive Detail":

> *The following types of excessively detailed textual information may be removed from UFSARs, except as indicated by applicable regulatory guidance or NRC Safety Evaluation Reports:*
>
> - *Descriptive information that is not important to providing an understanding of the plant's design and operation from either a general or system functional perspective, e.g., component model numbers*
> - *Design information that is not important to the description of the facility or presentation of its safety analysis and design bases, e.g., component details such as specific motor horsepower ratings for MOVs*
> - *Design information that, if changed during the life of the plant, would have no impact on the ability of plant systems, structures and components described in the UFSAR to perform their design basis function(s), e.g., specific HVAC equipment capacity and flow rate information for structures that do not contain equipment that performs design basis functions*
> - *Analytical information, e.g., detailed calculations, that is not important to providing an understanding of the safety analysis methodology, input assumptions and results, and/or compliance with relevant regulatory and industry standards*

Removal of excessive detail from UFSAR Section C3.6 that meets any of the above NEI 98-03 criteria would be permitted and documented using LS-AA-107-1001, Attachment 4, "Regulatory Review Form for a Non-Regulatory Change" with Site UFSAR Coordinator approval. Any changes that do not qualify as a non-regulatory change would require review under 10 CFR 50.59.

From interviews completed by one of the evaluators, it was confirmed that discussion of changes to the UFSAR had occurred. No interviewee identified any special direction being given to make a change. Any future change that is to be made regarding excessive detail would be evaluated by site Engineering and reviewed and approved by site Regulatory Assurance in accordance with EGC processes.

Detail 2: During a conference call on January 31, 2017, the petitioner pointed out that the UFSAR Section 3.6.2.1.2.1.2 is inconsistent with another section of the UFSAR in the use of the Break Exclusion Zone concept. Without discussion or review of the evidence supporting the position, the Byron manager dismissed the internal inconsistency by saying that the information supporting the position could be deleted as an UFSAR cleanup item.

Response:

UFSAR changes that resolve inconsistencies within the UFSAR are considered editorial changes and are documented using LS-AA-107-1001, Attachment 4 with Site UFSAR Coordinator approval. If the change was solely due to resolving an inconsistency within the UFSAR, LS-AA-107-1001, Attachment 4 would be used and a 50.59 Review would not be required. It should be noted that editorial changes are not addressed in NEI 98-03.

UFSAR Section 3.6.2.1.2.1.2 was last updated in UFSAR Revision 12 to "better describe the crack criteria for cracks in piping in the MSIV Rooms." As described in the associated 50.59 review (BRW-S-2007-183/6E-07-0109), the change to UFSAR paragraph 3.6.2.1.2.1.2.2.b was intended to clarify that the crack postulation criteria for Zone 1 is $0.8(1.2S_h + S_a)$ and for Zone 2 is $0.4(1.2S_h + S_a)$. Where the Zones are defined as follows:

Zone 1: Piping inside the isolation valve rooms from the containment wall to and including the containment isolation valve

Zone 2: Piping inside the isolation valve rooms from the containment isolation valve to the first restraint outboard of the containment isolation valve (which is the MSIV room wall)

Based on additional input from the petitioner, the petitioner believes there are inconsistencies between Sections 3.6.2.1.2.1.2.2.b and 3.6. Section 3.6.2.1.2.1.2, "Fluid System Piping in Containment Penetration Areas," applies to the fluid system piping inside the isolation valve rooms, which includes the main steamlines and the feedwater lines, starting at the inside of the containment wall and extending to the first restraint outside the containment isolation valve. Section 3.6.2.1.2.1.2.2.b was revised via DRP 12-030 to state the following:

b. Leakage cracks:

Per SRP 3.6.2, the break criteria of Subsection 3.6.2.1.2.1.2.2.a, paragraphs 1 and 2, also apply to the postulation of cracks in the penetration area in the region from the containment wall to and including the inboard or outboard isolation valves.

Leakage cracks in high energy ASME Section III Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping located in the

containment penetration area, other than that piping described in the paragraph above, are postulated in accordance with Subsection 3.6.2.1.2.1.1. For the Main Feedwater and Main Steam lines, this includes the piping from the inboard weld of the FWIV/MSIV to the first restraint outside the isolation valve (i.e., in the MSIV room wall).

Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Break of Piping," (page 3.6-2) states the following:

Areas of system piping where no breaks are postulated are:

a. *The main steam piping from the containment penetration fluid head outboard weld, to the upstream weld of the main steam pipe to the main steam isolation valve, including the main steam relief valve header and branch piping to the main steam power operated relief valve and main steam safety valves. This includes approximately 65 feet of piping (20 feet of header and 45 feet of relief piping) for each steam generator.*

b. *The main feedwater piping from the downstream weld of the main feedwater pipe to the main feedwater isolation valve, to the containment penetration fluid head outboard weld, including the main feedwater isolation valve bypass line from its branch off the main feedwater line to the upstream weld of the line to the normally closed feedwater backpurge isolation valve. This includes approximately 25 feet of piping for each steam generator.*

The UFSAR information in section 3.6 follows the format of Branch Technical Position MEB 3-1, attached to SRP 3.6.2. Section B.1.b states the following:

Fluid System Piping in Containment Penetration Areas

Breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves provided they meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the following additional design requiremetns...

Further, the UFSAR information in section 3.6.2.2.2.1.2.2 follows the format of BTP MEB 3-1 Section B.1.e which states:

With the exceptions of those portions of piping identified in B.1.b, leakage cracks should be postulated in ASME Code, Section III, Class 1 piping where the stress range by Eq. (10) of Paragraph NB-3653 exceeds 1.2 S_m, and in Class 2 and 3 or nonsafety class piping where the stress by the sum of Eq. (9) and (10) of Paragraph NC/ND 3652 exceeds 0.4 (1.2 S_h+ S_A). Nonsafety class piping which has not been evaluated to obtain similar stress information shall have cracks postulated at locations that result in the most severe environmental consequence.

The statements identified in the UFSAR correspond to the guidance in SRP 3.6.2 BTP MEB 3-1.

From interviews completed by one of the evaluators, it was confirmed that discussion of changes to the UFSAR had occurred. No interviewee identified any special direction being given to make a change. Any change that is to be made would be evaluated by site Engineering and reviewed and approved by site Regulatory Assurance, using appropriate processes.

Detail 3: Less than a month before the meeting discussed in Detail 2, there was an operability concern where Engineering management maintained a position of operability in the face of conflicting information.

Response:

From interviews of station personnel, it was identified that the operability concern was related to seismic qualification of relays. A determination of operability was made, after involving at least 28 staff members and subject matter experts at Byron Station, Braidwood Station, and Corporate. A review of documentation determined that the seismic qualification testing had been performed appropriately and results of the testing did not identify any operability issues.

Detail 4: Details 1 through 3, above, support a conclusion that Exelon management cherry picks information to support operability and dismisses contrary views.

Response:

ECP conducted interviews with personnel knowledgeable of the activities cited in the petition at Corporate and both Byron and Braidwood Stations. Interviews did not identify any concerns with respect to the environment for raising concerns, or with management dismissing contrary views. Interviews conclude that a healthy safety conscious work environment exists at Byron Station.

Supporting this evaluation, in November 2016, ECP performed an investigation into the health of the SCWE within Byron Engineering in response to an Issue Report documented in EGC's Corrective Action Program suggesting a degraded safety conscious work environment exists within the Engineering organization (IR 2738303). This investigation consisted of conducting interviews with 14 employees, primarily in Engineering. These interviews concluded that a healthy safety conscious work environment exists within the Engineering organization at Byron Station. In addition, ECP conducted pulsing interviews with 153 station personnel and observed 12 Engineering activities from December 2016 to present. Pulsing interviews are informal interactions with personnel used by ECP to gauge one's willingness to raise concerns, and to gauge the health of the safety conscious work environment. Included in the pulsing interviews were 17 Engineering personnel. Pulsings and observations revealed that workers are comfortable raising nuclear safety / quality concerns without fear of retaliation and further support the belief that the site culture does not tolerate retaliation for raising a nuclear safety / quality concern.

E. Conclusion

There was no evidence of any situation supporting a claim that EGC management cherry picks information to support operability. This is based on technical discussions being made in

collegial settings, and actions carried forth through established processes. Actions taken and behaviors demonstrated by EGC management in response to the issues / activities cited in the petition demonstrate a healthy Safety Conscious Work Environment exists.