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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

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

Subject: Response to Request for Additional Information Regarding Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, "Ultimate Heat Sink"

References:
1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" dated August 19, 2014 (ML14231A902)

2) Email from J. Wiebe (NRC) to J. Krejcie (Exelon Generation Company, LLC) Additional RAIs Regarding Containment Analysis for Braidwood UHS LAR (MF4671 and MF4672), dated July 22, 2015

3) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" dated November 9, 2015 (ML15313A254)

4) Letter from J. Wiebe (NRC) to B. Hanson (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 – Request for Additional Information Related to the License Amendment Request to Increase the Ultimate Heat Sink Temperature Limit," dated February 22, 2016 (ML16014A691)

In Reference 1, Exelon Generation Company, LLC, (EGC) requested an amendment to the Technical Specifications (TS) of Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2. The proposed amendment would modify TS 3.7.9, "Ultimate Heat Sink (UHS)," by changing the maximum allowable temperature of the UHS from 100°F to a maximum UHS temperature of 102°F. The U. S. Nuclear Regulatory Commission (NRC) requested additional information related to its review of Reference 1 and additional information was provided in various transmittals.

Subsequent to the submittal of Reference 1, the NRC requested additional information in regard to Braidwood Station's consideration of Westinghouse InfoGram 14-1 in Reference 2. Specifically, the NRC requested information related to the application of Westinghouse's InfoGram 14-1 described American Society of Mechanical Engineers (ASME) values for Reactor Coolant System (RCS) metal specific heat and density. EGC responded to the NRC request in Reference 3. However, the NRC notified EGC in Reference 4 that the Reference 3 response did not provide Braidwood Station, Units 1 and 2, plant-specific information.

The information contained in Attachment 1 is based on the results of plant-specific sensitivity analyses and studies performed to evaluate the impact of InfoGram 14-1 on the containment analyses and maximum UHS temperature. The sensitivity analyses and studies were only performed for the purpose of responding to the NRC request and are not used to revise the analysis of record or analyses as originally transmitted as part of Reference 1.

A clarification call was held with the NRC on April 20, 2016, to discuss the NRC's questions contained in Reference 4. During the teleconference, the NRC clarified that a response to NRC RAI-SCVB-15.f would require additional information to understand the InfoGram 14-1 impact on the maximum UHS temperature calculated in the LAKET-PC analysis (Attachment 5 of Reference 1). To that end, Westinghouse provided a sensitivity study for the maximum safeguards case for use in understanding the impact to the LAKET-PC model. The Westinghouse results supported a Sargent and Lundy sensitivity study for determining the impact on the maximum UHS temperature. The results of the sensitivity study discussed in the EGC response to NRC RAI-SCVB-15.f have been checked and verified to have been properly extracted from the computer files. The sensitivity analyses and studies were developed as information to be used to demonstrate the impact of InfoGram 14-1 on the containment analyses and LAKET-PC results; and are not intended to revise the Braidwood Station design basis.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Attachment 1 of Reference 1. The information provided in this letter does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

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

Should you have any questions concerning this letter, please contact Ms. Jessica Krejcie at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of April 2016.

Respectfully,

David M. Gullott
Manager - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachment 1: Response to Request for Additional Information

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

Attachment 1

Response to Request for Additional Information

ATTACHMENT 1
Response to Request for Additional Information

In Reference 1, Exelon Generation Company, LLC, (EGC) requested an amendment to the Technical Specifications (TS) of Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2. The proposed amendment would modify TS 3.7.9, "Ultimate Heat Sink (UHS)," by changing the maximum allowable temperature of the UHS from 100°F to a maximum UHS temperature of 102°F. The U. S. Nuclear Regulatory Commission (NRC) requested additional information related to its review of Reference 1 and additional information was provided in various transmittals.

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The following information provides the response to Reference 4 SCVB-RAI-15, parts a-f. The information below noted in italicized text denotes the NRC's quoted text.

SCVB-RAI-15

Exelon's response to SCVB-RAI-11 in its November 9, 2015, letter (ADAMS Accession No. ML 15313A254), did not provide Braidwood Station, Units 1 and 2, plant-specific information regarding the impact of the corrections resulting from Westinghouse InfoGram (IG) 14-1 on the loss-of-coolant accident (LOCA) mass and energy release, containment pressure, and temperature response, sump net positive suction head (NPSH) analysis, and minimum containment pressure for emergency core cooling system analysis. The Exelon response refers to analysis performed for a generic large containment plant which only addresses the impact on the containment peak pressure and temperature. The response does not provide a comparison of the related parameters of the generic plant with Braidwood Station.

Without additional quantitative information, the NRC staff evaluation of Braidwood Station, Units 1 and 2, containment performance due to increase in UHS temperature cannot be completed. After correcting the RCS metal specific heat and density to ASME values reported in IG 14-1, the NRC staff requests the following:

(a) Revise the analyses of record for the above containment analyses and provide results. As an alternative, by performing Braidwood Station, Units 1 and 2, plant-specific sensitivity analysis, provide increase in containment peak pressure, peak temperature, sump water temperature and decrease in the available NPSH for the pumps that draw suction flow from the sump during the LOCA recirculation phase.

EGC Response to SCVB-RAI-15.a:

Braidwood Station performed additional plant and unit specific sensitivity analyses to address this question. Braidwood Station Unit 1 and Unit 2 Loss Of Coolant Accident (LOCA) Mass and Energy (M&E) release analyses were re-performed utilizing the volumetric heat capacity calculated from the ASME material properties values as discussed in InfoGram 14-1. The LOCA M&E release analyses determined the initial stored metal energy in the RCS that is then used as an input into downstream containment analyses.

The volumetric heat capacity value used in the containment LOCA M&E release analyses is a calculated value. The bounding, constant volumetric heat capacity for the range of temperatures typically seen in the RCS, reactor vessel and steam generator secondary metal was determined to be 72.6 Btu/ft^3-°F. This value was used in the sensitivity analysis and replaces the original value of 53.2 Btu/ft^3-°F used in the Reference 1 analyses.

Only the volumetric heat capacity value was changed. The output of the LOCA M&E release analyses was used in the containment analyses.

The following table shows the results of the sensitivity analysis versus the results provided in the containment analyses in Reference 1:

Table 1a: Containment Peak Pressure

	Unit	Design Analysis (Pa) (psig)	UHS LAR Submittal (Reference 1) (psig)	Sensitivity Results (psig)	Increase from UHS LAR (psig)	Containment Design Limit (psig)
Double Ended Hot Leg Break (DEHL)	Unit 1	42.8	42.1	42.2	0.1	50
	Unit 2	38.4	37.7	37.8	0.1	50
Double Ended Pump Suction (DEPS)	Unit 1	42.8	42.1	43.8	1.7	50
	Unit 2	38.4	38.4	40.2	1.8	50

The sensitivity results show that while there is an increase in containment peak pressure, the increase is minor and within the containment design limit.

The sensitivity results for the peak containment temperature values from the containment analyses are given in Table 2a:

Table 2a: Containment Temperature					
	Break Location	Sensitivity Results (°F)	UHS LAR Submittal (Reference 1) (°F)	Increase (°F)	Equipment Qualification (EQ) Temperature Profile at time of Max Steam T (°F)
Unit 1	DEHL	263.62	263.50	0.12	325
	DEPS	265.24	262.60	2.64	270
Unit 2	DEHL	256.68	256.53	0.15	320
	DEPS	259.60	256.69	2.91	270

The calculated containment temperature profile from the sensitivity analysis has been reviewed against the containment temperature profile for EQ to determine the impact of increased containment temperature. Based on this review, the existing EQ profile remains bounding.

The results of the sensitivity analysis also determined the following for sump water temperature:

Table 3a: Sump Water Temperature				
	Break Location	Sensitivity Results (°F)	UHS LAR Submittal (Reference 1) (°F)	Change
Unit 1	DEHL	251.99	252.01	-0.02
	DEPS	261.43	258.14	3.29
Unit 2	DEHL	248.42	248.41	0.01
	DEPS	254.86	252.90	1.96

The results of the sump water temperature sensitivity analysis were used to determine the impact on the Containment Spray (CS) and Residual Heat Removal (RHR) pumps Net Positive Suction Head (NPSH) when the pumps take suction from the containment recirculation sumps. NPSH evaluation input parameters that could be impacted were reviewed and it was determined that adequate margin to the limits exists. Both the RHR and CS pumps still have positive NPSH margin considering the revised containment accident profiles.

The results of the NPSH sensitivity analysis are given below:

Table 4a: Net Positive Suction Head Evaluation Results			
Pump	UHS LAR Limiting Margin (Reference 1)	Limiting Margin from Sensitivity Analysis Results	Change
Residual Heat Removal	2.6 ft (at 248.4°F)	2.5 ft (at 258.1°F)	-0.1 ft
Containment Spray	1.3 ft (at 248.4°F)	1.3 ft (at 258.1°F)	0 ft

(b) In its November 9, 2015, response, Exelon stated that the containment peak pressure would increase for the Braidwood Station units. Discuss how this impacts the current integrated leak test rate and local leak rate test results.

EGC Response to SCVB-RAI-15.b:

The results of the sensitivity analysis were reviewed against the current Integrated Leak Rate Test (ILRT) and Local Leak Rate Test (LLRT) results. The results of the ILRT and LLRT still showed significant margin to the leakage limit when considering the increase in containment pressure from the sensitivity analysis.

The last ILRT containment pressure and measured leak rate after the stabilization period was compared to the expected results with the revised containment pressure from the sensitivity analysis. The impact of the resultant calculated peak pressures on the leak rate is estimated by applying an adjustment factor. Since leakage is proportional to the square root of the pressure, this factor is determined by taking the square root of the ratio of the new pressure over the old pressure. The results for each unit are shown in the tables below:

Table 1b: ILRT Braidwood Station

	ILRT Pressure (P_{ILT}) (psig)	Peak Sensitivity Analysis Pressure (P_{IG}) (psig)	Adjustment Factor $f_a = \sqrt{\frac{P_{IG}}{P_{ILT}}}$	Current Leak Rate (%/day)	Adjusted Leak Rate (%/day)	ILRT Leakage Limit (%/day)
A1R16 ILRT Unit 1	42.91	43.84	1.01	0.05571	0.056	0.15
A2R16 ILRT Unit 2	40.3819	40.16	No impact	0.1083	No impact	0.15

Table 1b shows that there is significant margin to the ILRT leakage limit, with consideration of the sensitivity analysis results. The Unit 2 ILRT results are not impacted because the peak containment pressure calculated in the sensitivity analysis is lower than the ILRT pressure.

The LLRT results for both Braidwood Unit 1 and Unit 2 are not affected because the actual test pressure for each of the tests is higher than the peak containment pressure that was calculated for the sensitivity analysis. Additionally, significant margin to the leakage rate limit exists as shown below.

The results of the review are summarized below:

Table 2b: LLRT Impact for Braidwood Station				
	Min Actual LLRT Pressure (psig)	Peak Sensitivity Analysis Containment Pressure (psig)	As-left Maximum Pathway Leakage Rate (scfh)	Leakage Rate Limit (scfh)
Unit 1	44.98	43.84	66.909	540.48
Unit 2	40.88	40.16	44.239	499.12

(c) Impact on the residual heat removal (RHR) heat exchanger heat load and on the discharge temperature of the closed cooling water (CC) from the RHR heat exchanger.

EGC Response to SCVB-RAI-15.c:

The results of the sensitivity analysis determined that the RHR heat exchanger heat load would increase by approximately 2% due to the calculated increase in sump water temperature (see response to RAI 15.a). The Component [Closed] Cooling Water (CC) heat exchanger removes the heat load from the RHR heat exchanger and is capable of removing the increased heat load.

Table 1c: RHR Heat Exchanger Heat Load			
	UHS LAR Analysis (Reference 1)	UHS Sensitivity Analysis	Change
Heat Load Rate (Btu/sec)	32,057.54	32,733.87	676.33 (2%)
Heat Load Rate (Btu/hr)	115,407,144	117,841,932	2,434,788 (2%)

Table 2c shows the new heat load for the CC heat exchanger and compares it to the limiting heat load calculated in support of the UHS LAR (Reference 1). The new heat load is significantly lower than the limiting heat load removal capability of the CC heat exchanger. The heat exchanger evaluation uses calculated fouling factors from the most recent performance tests, as described in Reference 5.

Table 2c: CC Heat Exchanger			
	UHS LAR Analysis (Reference 1)	UHS Sensitivity Analysis	Limiting Heat Removal
Heat Load (Btu/hr)	145.35×10^6	147.78×10^6	205.27×10^6

Additionally, the maximum UHS temperature of 105.2°F occurs at > 36 hours into the event. The expected heat load that the RHR heat exchanger and thus, CC heat exchanger would experience at that time is reduced significantly. Regardless, the CC heat exchanger design analysis conservatively uses an Essential Service Water (SX) inlet temperature of 106°F with the higher heat load from the containment analysis at < 1 hour into the event. The demonstrated capability to remove the required heat load (i.e., the results of the UHS sensitivity analysis) validates the results of the containment analysis.

The sensitivity analysis also calculated the impact on the CC return temperature from the RHR heat exchanger:

Table 3c: CC Return Temperature (from RHR heat exchanger)			
	UHS LAR Analysis (Reference 1) (°F)	UHS Sensitivity Analysis (°F)	Change (°F)
CC Temperature (°F)	179.19	180.69	1.5

The CC pump NPSH has been evaluated at the higher temperature and has been found to remain acceptable.

(d) Impact on the reactor containment fan cooler (RCFC) heat load and on the exit temperature of the essential service water (SX) from the RCFC.

<u>EGC Response to SCVB-RAI-15.d:</u>

The sensitivity analysis determined the heat removal from each Reactor Containment Fan Cooler (RCFC) in the containment analysis and the impact to the heat removal capability due to the change in volumetric heat capacity value. This heat removal rate is compared to the UHS LAR submittal value.

Table 1d: Heat Removal - RCFC			
	UHS LAR (Reference 1)	Sensitivity Analysis	Change
Unit 1 – Maximum Heat Removal per RCFC (Btu/sec)	31,649.03	32,336.09	687 (+ 2.2%)
Unit 2 –Maximum Heat Removal per RCFC (Btu/sec)	30,283.46	30,929.13	646 (+ 2.1%)

The results showed an approximate 2% change in required heat removal. Refer to the response to SCVB-RAI-15.a for the resulting impact to the containment analysis (i.e., peak containment pressure and peak containment temperature).

The SX water discharge temperature is not evaluated on a component specific basis for overall impact to the maximum UHS temperature. For the impact on the exit temperature of SX from the RCFCs, refer to the response to SCVB-RAI-15.f.

(e) Overall impact on the temperature of SX exiting the CC heat exchanger.

EGC Response to SCVB-RAI-15.e:

The SX discharge temperature is not evaluated on a component specific basis for overall impact to the UHS temperature. The method for calculating the temperature rise across the plant for the SX water is shown in Section 2.1.3 of the UHS temperature analysis (Attachment 5 of Reference 1). For the impact on the exit temperature of SX from the CC heat exchangers, refer to the response to SCVB-RAI-15.f.

(f) Overall impact on the temperature of SX returning to the UHS.

EGC Response to SCVB-RAI-15.f:

The impact of InfoGram 14-1 on the limiting heat loads for the containment analyses and maximum UHS temperature were reviewed. The two analyses have different purposes and utilize different inputs which are conservatively selected for the purpose of ensuring that each specific analysis results are conservative and bounding.

Specifically, the containment analysis minimizes heat removal assumed from the various components to maximize and conservatively calculate containment peak pressure and temperature. For example, two of the available four RCFCs are assumed to be in operation and the heat removal curve for the SX system at 104°F is used to minimize heat removal and thus maximize containment pressure and temperature. The results of the sensitivity analysis for the impact of InfoGram 14-1 on the containment analysis are described in the response to SCVB-RAI-15.a.

In contrast, the analysis that determines the heat load to the UHS utilizes inputs that are selected to maximize heat input to the UHS and thus, maximize the resulting UHS temperature. For the RCFCs, the analysis assumes all four RCFCs are in operation and uses a heat removal curve for SX at 95°F.

The analysis that calculated maximum heat load to the UHS was reviewed and contains several conservative assumptions. To address the expected increase in heat load due to InfoGram 14-1, conservatisms within the existing model were identified and adjusted. Specifically, the containment sump recirculation and SX inputs (e.g., flow rates and heat removal curves) were adjusted. Additionally, similar to the Reference 1 containment analysis, recent Nuclear Safety Advisory Letters (NSALs) were addressed by modifying inputs as appropriate. The model was then rerun to assess the impact of changes in the volumetric heat capacity value.

The results of the sensitivity study show that the combined heat removal rates for the RCFC and the RHR heat exchangers remain below the values from the maximum heat load analysis (used as input to Attachment 5 of Reference 1) except for the time period from 1899 seconds thru 9999 seconds following the accident. During the time period from 1899 seconds through 9999 seconds, heat load values increased slightly.

Based on the increased heat load, an increase in the integrated heat load of 2% for the 0-3 hour time segment and 0.1% for the 3-6 hour time segment was conservatively assumed and input to the UHS temperature analysis. The LAKET-PC model was rerun (consistent with Attachment 5 of Reference 1) to determine the impact to the maximum UHS temperature, and thus, the maximum SX temperature. No credit for the reduced heat load during other time periods (i.e., time period before 1899 and after 9999 seconds) was assumed. The results demonstrated a peak UHS temperature of 105.3°F with the increased heat load. This temperature is acceptable since the component specific equipment evaluations were performed using 106°F as the maximum post-accident temperature. Use of 106°F in the equipment calculations is discussed in the Reference 5 transmittal.

An additional sensitivity case was run for the maximum UHS temperature analysis. It was determined that the heat load to the UHS would need to increase by approximately 20% in order to challenge the maximum post-accident temperature assumed in the component specific evaluations of 106°F. Therefore, the analyses supporting the increase in the maximum allowable UHS temperature in TS 3.7.9 to 102°F contain conservatism to address the impact of consideration of InfoGram 14-1.

ATTACHMENT 1
Response to Request for Additional Information

References:

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" dated August 19, 2014 (ML14231A902)

2) Email from J. Wiebe (NRC) to J. Krejcie (Exelon Generation Company, LLC) Additional RAIs Regarding Containment Analysis for Braidwood UHS LAR (MF4671 and MF4672), dated July 22, 2015

3) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" dated November 9, 2015

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5) Letter D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Supplemental Information Regarding Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" dated February 12, 2016 (ML16043A496)