

10 CFR 50.90

RS-12-033

February 20, 2012

U. S. Nuclear Regulatory Commission
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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: Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate

- References:**
1. Letter from Craig Lambert (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated June 23, 2011
 2. Letter from B. Mozafari (U. S. NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Units 1 and 2 – Request for Additional Information RE: Measurement Uncertainty Recapture Power Uprate Request (TAC NOS. ME6587, ME6588, 6589, AND ME6590)," dated February 14, 2012 [ML 120270146]
 3. Letter from B. Mozafari (U. S. NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2 and Braidwood Station, Units 1 and 2 – Request for Additional Information RE: Measurement Uncertainty Recapture Power Uprate Request (TAC NOS. ME6587, ME6588, 6589, AND ME6590)," dated February 14, 2012 [ML 120260936]

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications to implement an increase in rated thermal power of approximately 1.63% based on increased feedwater flow measurement accuracy. In References 2 and 3, the NRC requested additional information to support review of the proposed changes. In response to this request, EGC is providing the attached information for all of the requests with the exception of the Civil and Mechanical Branch [ECMB] Request 13 in Reference 2 and the Balance of Plant Branch [SBPB] Request 1 in Reference 3. EGC will be

providing the response to these two requests under separate transmittal as indicated in Attachment 1.

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

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Leslie E. Holden at (630) 657-3316.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of February 2012.

Respectfully,



Kevin F. Borton
Manager, Licensing - Power Uprate

Attachment 1: Response to Request for Additional Information

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

**Braidwood and Byron Stations
Measurement Uncertainty Recapture License Amendment Request (MUR LAR)**

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
February 20, 2012**

ATTACHMENT 1

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

(NON-PROPRIETARY)

NON-PROPRIETARY

NRC/Mechanical and Civil Engineering Branch (EMCB)

NRC/EMCB Request 1

Section IV.1.A.ii.f of Attachment 7 to the license amendment request (LAR) discusses the structural evaluation of the lower and upper core support assemblies for the effects of increased heat generation rates. Provide further information and confirm that

- a. the proposed MUR power uprate only affects the design loads associated with heat generation rates and all other design loads associated with the design of the reactor vessel internals are unaffected by the proposed MUR power uprate;*
- b. all design loading conditions, as noted in Section 3.9.5.2 of the Byron and Braidwood updated final safety analysis report (UFSAR), were considered in the structural re-evaluation of the reactor vessel internal components to assess the impact of the proposed MUR power uprate; and*
- c. the original design codes of record were utilized in the structural re-evaluation of the reactor vessel internal components.*

Provide the maximum calculated stresses and cumulative fatigue usage factor for the most limiting component of the reactor vessel internals and their respective comparison with the Byron and Braidwood design basis acceptance criteria.

Response

The Byron and Braidwood reactor vessel internal components analysis of record (AOR) was performed with conservative gamma heating rates. The Measurement Uncertainty Recapture (MUR) power uprate gamma heating rates were verified to remain bounded by the conservative heating rates used in the AOR.

All the design loading conditions noted in Section 3.9.5.2 of the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR) were considered in the structural assessment of the reactor vessel internal components to assess the impact of the proposed MUR power uprate. The design loads associated with the design of the reactor vessel internals remain bounded by the AOR.

The Byron and Braidwood Units 1 and 2 reactor vessel internals components were designed prior to the introduction of Subsection NG of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, and are not licensed to meet any specified edition or addenda of the ASME Code. As a result, a plant-specific stress report of the reactor internals was not required. However, the design of the reactor internals is evaluated according to the Westinghouse Criteria which is similar to the criteria described in the Subsection NG of the ASME code. The Westinghouse acceptance criteria are the same as those used in the original design of the plant and its original licensing basis.

The maximum calculated stresses and cumulative fatigue usage factor for the most-limiting component of the reactor vessel internals are unaffected by the MUR power uprate and remain bounded by the AOR.

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NRC/EMCB Request 2

Section 3.9.5.1 of the Byron and Braidwood UFSAR describes the reactor vessel internals as three parts consisting of the lower core support structure, the upper core support structure, and the incore instrumentation support structure. Section IV of Attachment 7 to the LAR does not discuss the incore instrumentation support structures. Provide further information relative to the impact of the design conditions associated with the proposed MUR power uprate on the incore instrumentation support structures.

Response

As stated in UFSAR Section 3.9.5.1, the in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The proposed MUR power uprate impact on the incore instrumentation support structures, including both the upper support columns and the lower support columns was assessed. Since the current analyses loads (i.e. LOCA hydraulic forces and seismic loads) are not changing from the current analysis of record and remain bounded for the MUR power uprate, the stresses and the cumulative fatigue usage factors in these components remain unchanged from the current analysis of record.

NRC/EMCB Request 3

Provide further information and confirm that, for the proposed MUR power uprate conditions, the maximum deflection values allowed for the reactor vessel internal support structures, as noted in Table 3.9-4 of the Byron and Braidwood UFSAR, are maintained.

Response

The design inputs, i.e. LOCA hydraulic and seismic forces and geometry, are not changing from the current analysis of record for the MUR power uprate; therefore, there is no impact on the allowable deflections provided in Byron and Braidwood UFSAR Table 3.9-4, "Maximum Deflections Allowed for Reactor Internal Support Structure." The values provided in UFSAR Table 3.9-4 remain valid for the MUR power uprate.

NRC/EMCB Request 4

Section IV.1.B.iv.1 of Attachment 7 to the LAR states that there is an approximate 1.2°F increase in temperature difference across the core (T_{hot} increases approximately 0.6°F and T_{cold} decreases approximately 0.6°F) from current operating conditions due to the MUR power uprate. Section IV.1.A.i of Attachment 7 to the LAR discusses reactor vessel structural evaluation and states that due to operational restrictions, the MUR minimum vessel inlet and maximum vessel outlet temperatures are limited to 538.2°F and 618.4°F, respectively. Provide further clarification on temperature effects relative to the values in Tables 3-1 and 3-2 of Attachment 1 to the LAR, the statements in Sections IV.1.B.iv.1 and IV.1.A.i of the LAR, and the temperatures used in the analysis of record.

Furthermore, the lifting lug loads and evaluation are discussed in Section IV.1.A.i of Attachment 7 to the LAR. The terminology of "lifting lug" and its relation to and its inclusion in the proposed MUR power uprate license amendment is not clear. Provide further information to clarify which

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reactor vessel component is referred to as "lifting lug". Also, regarding the affected reactor vessel component,

- a. provide a table summarizing the comparison of design parameters for the current operation conditions, MUR power uprate conditions, and design basis conditions; and
- b. provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location of the affected component and their respective comparison with the Byron and Braidwood design basis acceptance criteria.

Response

The MUR power uprate Reactor Coolant System (RCS) design conditions given in Tables 3-1 and 3-2 provide a T_{avg} range in which the minimum T_{cold} is 541.4°F and the maximum T_{hot} is 620.9°F. The reactor vessel analysis of record (AOR) evaluated a minimum T_{cold} of 538.2°F and a maximum T_{hot} of 620.3°F. Therefore, the MUR power uprate maximum T_{hot} of 620.9°F exceeds the maximum T_{hot} evaluated in the reactor vessel AOR. Note that the MUR power uprate minimum T_{cold} is bounded by the minimum T_{cold} evaluated in the reactor vessel AOR. Normally, a reconciliation would be necessary because the MUR power uprate maximum T_{hot} is not bounded by the maximum T_{hot} evaluated in the reactor vessel AOR. However, all Byron and Braidwood units have plant operational limits which restrict the minimum T_{cold} to 538.2°F and the maximum T_{hot} to 618.4°F. The plant operational limits will remain in place for the MUR power uprate. Therefore, the minimum T_{cold} and maximum T_{hot} evaluated in the reactor vessel AOR bound those of the MUR power uprate when the plant operational limits are taken into consideration.

There are three lifting lugs oriented 120° apart around the external side of the reactor vessel closure head. The Integrated Head Package (IHP) lift rod assemblies attach to the lifting lugs through a lift rod clevis and clevis pin. Figures EMCB R4-1 and R4-2 depict how the lifting lugs are attached to the reactor vessel closure head.

The lifting lug mechanical loads identified for current operating conditions did not change due to the MUR power uprate.

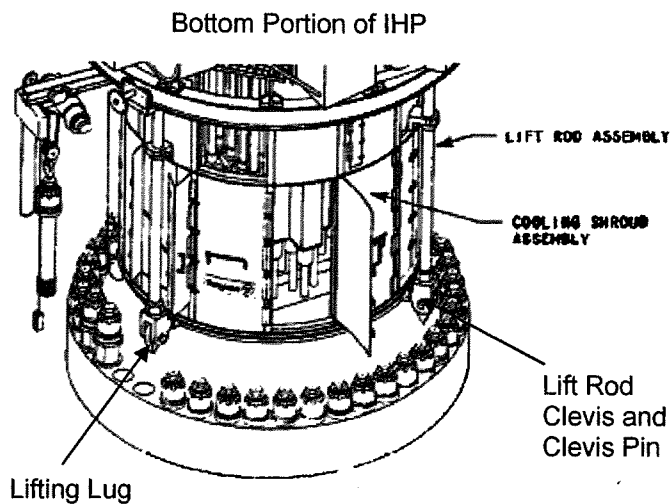


Figure EMCB R4 - 1: Bottom Portion of Integrated Head Package to Reactor Vessel Closure Head

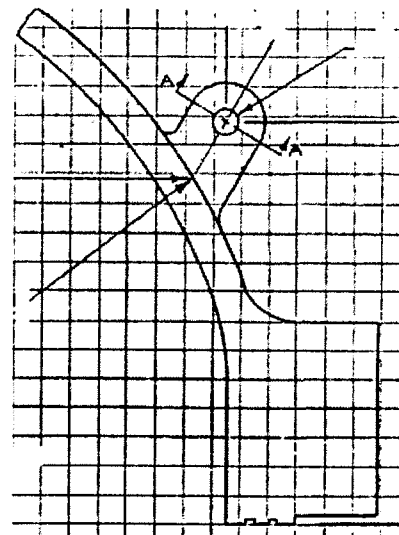


Figure EMCB R4 - 2: Detail of Lifting Lug Attachment to Reactor Vessel Closure Head

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NRC/EMCB Request 5

Section IV.1.A.iii of Attachment 7 to the LAR discusses the control rod drive mechanism (CRDM). In this section, it is stated that updated seismic and loss-of-coolant accident (LOCA) loads remain less than the allowable loads provided in the analysis of record. This statement implies that the seismic loads have been updated. Also, this statement is not consistent with Section IV.1.A.ii.e of Attachment 7 to the LAR where it is stated that the proposed MUR power uprate conditions do not affect the current design basis for seismic and LOCA loads. Provide further clarification.

Furthermore, Section IV.1.A.iii of Attachment 7 to the LAR states that CRDM is subjected to cold temperatures and reactor coolant system pressures and these are the only design parameters considered in the CRDM evaluation. Elaborate and confirm that:

- a. the design basis loading conditions and operational requirements, as noted in Section 3.9.4 of the Byron and Braidwood UFSAR, have been considered in the structural evaluation of the control rod drive system for the proposed MUR power uprate conditions; and*
- b. the control rod drive system will continue to be in compliance with the Byron and Braidwood design basis acceptance criteria under the proposed MUR power uprate conditions.*

Response

A seismic and loss of coolant accident (LOCA) loads assessment was completed as part of the MUR power uprate. The assessment concluded that MUR uprate conditions have no impact on the seismic/LOCA loads and the existing seismic/LOCA loads remain valid and unchanged for the MUR power uprate.

The CRDM assessment completed for the MUR uprate project considered all pressure and thermal design transients and load combinations noted in Section 3.9.4 of the Byron Braidwood UFSAR. The CRDM assessment concluded that the pressure and thermal design transients due to the MUR uprate have no impact on the CRDM qualification analyses of record. The CRDM qualification analyses of record demonstrated that Byron and Braidwood are in compliance with the ASME Code stress criteria.

NRC/EMCB Request 6

Provide further information and confirm that the design basis pressure and temperatures (normal operating and accident temperatures) used in the design of the containment structure, including the steel liner plate, and its internal structures remain bounding following the proposed MUR power uprate.

Response

The design basis containment pressure and temperature for normal operation are delineated respectively in Byron/Braidwood Technical Specification 3.6.4 and 3.6.5. Assessments performed for the MUR power uprate concluded that these normal operation design parameters remain applicable.

Accident containment parameters were evaluated for the MUR power uprate. For primary system pipe breaks (i.e., LOCAs), as discussed in the MUR LAR submittal (Reference 1), Section III.15.5, "LOCA Long Term Mass and Energy Release and Containment Response –

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UFSAR 6.2.1.3.1, Analysis Results,” the containment peak pressure and temperature for the MUR remain bounded by the containment structure design pressure and temperature with margin.

For secondary pipe breaks (Main Steam Line Breaks (MSLB)), as discussed in the MUR LAR submittal (Reference 1), Section III.16.5, “Main Steam Line Break Mass and Energy Releases Inside Containment – UFSAR 6.2.1.4, Analysis Results,” the peak pressure remains bounded by the containment design pressure with margin and there is a very small calculated increase (+0.6°F) in the peak containment air temperature for Unit 1. Unit 2 remains bounded by the analysis of record.

Exelon’s response (Reference 2) to the NRC Request for Additional Information (Reference 3) Request 10, summarized the temperatures and pressures from the LOCA and MSLB Mass and Energy Analyses for Byron/Braidwood MUR.

As discussed in the UFSAR Section 6.2.1.1.3, “Containment Structure, Design Evaluation,” the justification for the design temperatures selected for the liner and internal containment structures is that they are conservative when the duration of the peak temperature for the secondary side (i.e., steam line) break, the temperature lag between the containment atmosphere and the passive heat sinks such as the containment liner and internal structures, and the resistance to heat transfer provided by the materials used, are considered. This justification remains applicable for MUR power uprate because the duration remains short. Figure 10-1, “Containment EQ Temperature and Pressure Profile,” in Reference 2 shows that the MSLB temperature profile for the MUR power uprate falls below the containment design temperature of 280°F less than 200 seconds after the onset of the MSLB.

The assessment performed for the MUR power uprate indicated that the structural effect of the MSLB temperature on the containment structure remains bounded by the LOCA case. Therefore for both units the containment structure remains acceptable for both primary and secondary system pipe breaks.

For the containment internal structures, RCS initial pressure and temperature for MUR were reviewed and confirmed to be bounded by the inputs to the existing short-term LOCA mass and energy releases. Therefore the containment internal structures remain acceptable for the MUR power uprate.

NRC/EMCB Request 7

Section IV.1.A.iv “Reactor Coolant Piping and Supports” of Attachment 7 to the LAR discusses the effects of the proposed MUR power uprate mostly on a qualitative basis and the term “no significant changes” has been used in several areas to describe the impact of the proposed MUR power uprate. Discuss in more detail the information relative to the revised design conditions, before and after the proposed MUR power uprate, for those components evaluated under Section IV.1.A.iv of Attachment 7 to the LAR.

Summarize the results of any additional evaluations performed for the affected components and indicate whether these components remain bounded by the current analysis of record. For those components that were not bounded by the analysis of record:

- a. provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location; and*
- b. provide further clarification that the re-evaluation was performed in accordance with the design basis code of record and the affected components continue to remain in compliance with the Byron and Braidwood stations design basis acceptance criteria.*

NON-PROPRIETARY**Response**

The conditions associated with the MUR power uprate were evaluated to determine the impact on the existing as-built design basis reactor coolant loop (RCL) analysis for the following:

- RCL piping stresses and displacements,
- Primary equipment nozzle loads (reactor pressure vessel (RPV) inlet and outlet nozzles, steam generator inlet and outlet nozzles, and reactor coolant pump (RCP) suction and discharge nozzles),
- Primary equipment support loads (RPV nozzle supports, steam generator columns and lateral bumpers, RCP columns and lateral supports, and pressurizer supports), and
- Pressurizer surge line piping stresses and displacements including the effects of thermal stratification.

The following inputs were considered in the assessment:

- Nuclear Steam Supply System (NSSS) Design Parameters,
- NSSS design transients,
- Loss-of-coolant accident (LOCA) hydraulic forcing functions loads, and
- RPV motions due to LOCA.

The RCL piping assessment for the MUR power uprate was performed in accordance with the Byron/Braidwood design basis code of record (ASME, Section III, 1974 Edition, including Summer 1975 addenda).

The RCL thermal, deadweight, seismic, fatigue, LOCA and Main Steam / Feedwater line break analyses were reconciled to the design inputs as follows:

RCL Thermal Analysis

The RCL piping in the existing design basis was evaluated for the conditions associated with a RCS hot leg upper bound temperature of 618.4°F, cross-over leg temperature of 555.4°F, and a cold leg temperature of 555.7°F. The reactor coolant upper bound temperatures for the MUR power uprate did not increase for the hot leg, they decreased by 0.6°F for the cross-over leg, and they decreased by 0.6°F for the cold leg as compared to the current design basis temperatures. The MUR power uprate upper bound thermal NSSS design parameters are bounded by the design basis analysis.

Considering the RCL MUR power uprate lower bound temperature case, there is a temperature operating window as follows: 9.8°F between the upper bound T_{high} and lower bound T_{low} for the hot leg, 16.9°F between the upper bound T_{high} and lower bound T_{low} for the cross-over leg, and 16.9°F between the upper bound T_{high} and lower bound T_{low} for the cold leg.

The thermal piping stresses and displacements are dependent on the coefficient of thermal expansion and temperature difference between ambient to hot conditions. The coefficient of thermal expansion increases with an increase in temperature. The thermal piping loads and thermal stresses for the lower bound temperatures are lower than the corresponding loads and stresses for the upper bound case. Therefore, the thermal stresses for the upper bound case are higher, and the upper bound case piping stresses, primary equipment nozzle loads, primary equipment support loads (including the reactor vessel, steam generator, reactor coolant pump and pressurizer), and the auxiliary line displacements at the connections to the RCL are limiting.

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Since there is no increase in upper-bound temperature in comparison to the hot leg, cross-over leg, and cold leg temperatures in the current RCL thermal analysis design basis, the current RCL thermal analysis design basis analysis remains bounding.

RCL Deadweight and Seismic Analysis

There is no change in deadweight because there is no change to the configuration of the RCL piping and supports due to the MUR power uprate. The seismic response spectrum does not change due to the MUR power uprate. Therefore, it is concluded that there are no changes to RCL deadweight and seismic analyses for the MUR power uprate.

RCL Fatigue and Surge Line Stratification

There are no changes to the primary side NSSS design transients due to the MUR power uprate. Also, the pressurizer surge line transients do not change. Therefore, there is no impact on the piping for the MUR power uprate due to the NSSS design transients. There is no adverse effect on the fatigue evaluation of the RCL and pressurizer surge line, including the effects of thermal stratification. The pressurizer surge line stratification analysis continues to meet the code of record (ASME, Section III, 1986 Edition).

LOCA Analysis

The impact on the LOCA hydraulic forcing functions (HFFs) due to the MUR power uprate has been assessed for the accumulator and surge line breaks. Based on this assessment, the LOCA HFFs used in the existing RCL piping LOCA analyses remains bounding for the MUR power uprate.

The impact on the RPV motions due to MUR power uprate has been assessed. Based on this assessment, the LOCA RPV motions used in the existing RCL piping LOCA analyses remains bounding for the MUR power uprate.

Main Steam and Feedwater Line Break

The design basis main steam and feedwater line break analyses remain valid for the MUR power uprate. Based on the NSSS design parameters, the main steam line and feedwater line break pressures decrease and the feedwater temperature decreases slightly for the MUR power uprate. A decrease in pressure will reduce the thrust and jet impingement forces; however a decrease in temperature may increase the forces due to fluid momentum. These small differences will offset each other such that the thrust and jet impingement forces used in the current analysis remain bounding.

Based on the above, there are no changes due to the MUR power uprate to the piping or component qualification from the design basis, including: primary equipment nozzles and supports, Class 1 auxiliary piping analysis, and surge line stratification. The maximum primary and secondary stresses and the maximum fatigue usage factors associated with the existing design basis analysis are applicable to the MUR power uprate. The above components continue to remain in compliance with the Byron/Braidwood design basis acceptance criteria.

NRC/EMCB Request 8

Section IV.1.A.v of Attachment 7 to the LAR discusses the evaluation of balance of plant (BOP) piping systems. Confirm that other BOP piping systems (e.g., chemical and volume control, auxiliary feedwater, fuel pool cooling, containment spray, essential service water, safety injection) that may be affected by the MUR uprate conditions have been evaluated and provide a complete list of BOP piping systems evaluated in support of MUR power uprate. Discuss the

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methodology used for evaluating BOP piping, including pipe supports, and provide further information relative to the design conditions in each BOP piping system, before and after the proposed MUR power uprate. Summarize the results of the additional evaluations performed for the affected piping systems and indicate whether these piping systems remain bounded by the current analysis of record. For those BOP piping systems not bounded by the current analysis of record:

- a. *provide the maximum calculated stresses and cumulative fatigue usage factors at the most critical location in each unbounded piping system; and*
- b. *provide further clarification that the re-evaluation of the piping system, including pipe supports, was performed in accordance with the design basis code of record and in compliance with the Byron and Braidwood stations design basis acceptance criteria.*

Furthermore, state whether any piping or pipe support modifications are required to support the proposed MUR power uprate.

Response

The following Byron and Braidwood Stations Balance of Plant/Nuclear Steam Supply System (BOP/NSSS) piping systems were assessed for MUR power uprate conditions:

- Main Steam System
- Extraction Steam System
- Condensate System
- Condensate Booster System
- Heater Drains System
- Feedwater System
- Steam Generator Blowdown System
- Auxiliary Steam System
- Auxiliary Feedwater System
- Chemical and Volume Control System
- Fuel Pool Cooling System
- Safety Injection System
- Essential Service Water System
- Component Cooling Water System
- Containment Spray System
- Non-Essential Service Water
- Circulating Water

It was determined that the following Byron and Braidwood Stations BOP/NSSS piping systems are not negatively impacted (i.e., an increase in temperature or pressure) by MUR power uprate:

- Steam Generator Blowdown System
- Auxiliary Feedwater System
- Chemical and Volume Control System
- Safety Injection System
- Containment Spray System
- Circulating Water

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For these piping systems no further assessment was performed. These systems remain bounded.

For the remaining systems (i.e., those that were assessed to have an increase in temperature and/or pressure) the methodology and acceptance criteria discussed in the paragraphs below were applied to assess the acceptability of the piping for the MUR power uprate.

Operating pressures and temperatures in each line under Current Licensed Thermal Power (CLTP) and MUR power uprate were reviewed against the design pressure and temperature of the line.

For non-seismic piping, the increase in pressure was considered to be acceptable provided that the MUR power uprate operating pressure was bounded by the design pressure. As a result of the MUR power uprate, there were no non-seismic systems that exceeded the design pressure. For seismic piping, there were no pressure increases as a result of the MUR power uprate.

The increase in temperature was considered to be acceptable provided that the MUR power uprate operating temperature did not increase by more than 1% compared to CLTP operating temperature or the MUR operating temperature remained less than 150° F. For lines that are currently qualified to be within Code thermal stress allowable, increasing the system temperature range by <1% will not affect the acceptability of the piping/support system. Decreasing the system temperature will increase the allowable stress margin. For evaluating pipe thermal expansion stress, the temperature range is equal to the maximum operating temperature minus the normal ambient temperature, or 70°F. This represents the largest change in temperature that the pipelines can experience. Typically, pipe thermal stress is not evaluated for operating temperatures less than 150°F.

For piping segments which do not pass the screening criterion (i.e., <1% change), a detailed review of pipe stress calculations is conducted to determine if margin exists to accommodate thermal expansion stresses at MUR power uprate.

All of the systems, except for the heater drain piping and condensate booster piping are considered to remain bounded based on the above criteria. The heater drain system piping experiences a maximum temperature increase of 1.43%. The design basis analysis was found to bound the MUR condition because the design basis analysis used operating temperature of 187°F while the CLTP operating temperature is 160.8°F and the MUR operating temperature is 162.1°F temperature. The condensate booster piping experiences a maximum temperature increase of 1.10%. The design basis analysis was found to bound MUR conditions because the design basis analysis used an operating temperature of 176°F while the CLTP operating temperature is 161.0°F and the MUR operating temperature is 162.0°F. Therefore, the BOP/NSSS piping systems are considered to remain in compliance with their current design basis code of record and the Byron and Braidwood stations design basis acceptance criteria.

Since there were no significant increases in piping temperatures, pipe support loads did not experience an appreciable increase. Therefore, no pipe or pipe support modifications are required for MUR power uprate conditions.

NRC/EMCB Request 9

Section IV.1.A.viii of Attachment 7 to the LAR discusses the pressurizer structural evaluation. In this section of the LAR, it is stated that the revised design parameters have an insignificant impact on the fatigue analysis results. It is also stated that the proposed MUR power uprate has a negligible impact on the qualification of the pressurizer surge, spray, safety and relief nozzle structural weld overlay designs. Provide further information to support the above qualitative

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statements and to demonstrate compliance with the Byron and Braidwood design basis acceptance criteria. Also, provide a table summarizing the comparison of pressurizer design parameters for the current operation conditions, MUR power uprate conditions, and design basis conditions.

Response

Heat-up of the pressurizer from the cold condition to the hot standby condition is independent of plant power level and is unaffected by an uprate which may affect RCS temperatures and transients between hot standby and 100% power operation. The pressurizer maintains the RCS pressure and provides a cushion to accommodate changes in fluid volume and provides overpressure protection to the RCS. The temperature within the pressurizer is at the saturation temperature. Therefore, transients that will affect the fatigue analysis for pressurizer components are the result of changes to the fluid temperature entering the pressurizer, i.e., insurge/ outsurge through the surge line or spray through the spray line, or as a result in changes to the transients affecting the pressurizer pressure transients. Previous Westinghouse evaluations of design transients following an MUR power uprate show that the only transients that are affected are those that are the result of the feedwater changes and affect only the steam generator secondary side components. There are no transients affected that pertain to the pressurizer, temperature or pressure. Therefore, there is no impact on the pressurizer analysis as a result of MUR power uprate transient changes. Given that the transients are unchanged, the impact on the lower pressurizer components due to insurge/outsurge and the upper pressurizer components due to spray will change only if the temperature of the fluid changes, and then only if the temperature change increases. For this to happen, the RCS temperature for T_{hot} , affecting insurge/outsurge, and T_{cold} , affecting the spray temperature, would have to decrease from the analyzed condition.

The Table EMCB R9-1 provides a comparison showing the temperature change across the pressurizer components evaluated for the design basis conditions, the current operating conditions, and at MUR power uprate conditions. It is seen from Table EMCB R9-1 that the temperature change for T_{hot} , affecting the lower pressurizer (ΔT_{hot}), is less at MUR power uprate conditions by 0.6 °F and is enveloped by the analysis of record (AOR). The temperature differential for the upper portion of the pressurizer is shown to exceed the current operating condition by 0.6 °F (ΔT_{cold}). This is an increase of approximately 0.5% over the current operating condition ΔT_{cold} and is not considered to be significant.

Also, since the baseline analysis, which is also the AOR, continues to envelope the MUR power uprate temperature differential, the AOR is not affected and remains applicable. Therefore, there is no change to the baseline analysis results due to the MUR power uprate resulting from changes to the RCS temperatures affecting the pressurizer.

An assessment of the pressurizer surge, spray, safety and relief nozzle for structural weld overlay (SWOL) was also performed as part of the MUR power uprate. The assessment concluded that the MUR power uprate would have no impact on the AOR for these components based on the findings previously noted. Therefore, the MUR power uprate is enveloped by the current SWOL analysis and is acceptable.

NON-PROPRIETARY

Parameter	Baseline Analysis (AOR) (°F)	Current Operating Conditions (°F)	MUR Operating Conditions (°F)
$T_{\text{pressurizer}}$	652.7	652.7	652.7
T_{hot}	542.7	608	608.6
T_{cold}	517.7	542	541.4
$\Delta T_{\text{hot}} = T_{\text{pressurizer}} - T_{\text{hot}}$	110	44.7	44.1
$\Delta T_{\text{cold}} = T_{\text{pressurizer}} - T_{\text{cold}}$	135	110.7	111.3

NRC/EMCB Request 10

Section IV.1.B.iii of Attachment 7 to the LAR discusses the evaluation of the reactor vessel internal components for flow induced vibration (FIV) impact under MUR power uprate conditions. Also, Section IV.1.A.ii.e of Attachment 7 to the LAR states that the FIV stress levels on the core barrel assembly and upper internals are below the material high-cycle fatigue endurance limit and the proposed MUR uprated conditions do not affect the structural margin for FIV. Provide further information relative to those design parameters, before and after MUR power uprate, which could potentially influence FIV response of the reactor internals. Also, discuss the comparison of alternating stress intensities to design basis allowable limits for the most critical components demonstrating compliance with the Byron and Braidwood design basis acceptance criteria.

Response

Comparisons of flow induced vibration (FIV) design parameters before and after the MUR power uprate are provided in Table EMCB R10-1.

Parameter	Current Analysis of Record	MUR Power Uprate	Ratio
Mechanical Design Flow (gpm/loop)	107,000	107,000	1.0
Vessel Inlet Temperature (°F) / Fluid Density (lbm/ft ³)	542 / 47.369	541.4 / 47.385	~1.0
Vessel Outlet Temperature (°F) / Fluid Density (lbm/ft ³)	608 / 42.4535	608.6 / 42.411	~1.0

The MUR power uprate design conditions will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces, induced by flow. The corresponding T_{cold} and T_{hot} fluid densities change by less than 0.1% from the current analyzed condition. Therefore, the effect on the flow-induced vibration stresses (alternating stress intensities) due to MUR power uprate on the reactor internals remains unchanged from the current analysis of record.

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Discuss further information and confirm that the nuclear steam supply system component supports, as discussed in Section 3.9.3.4 of the Byron and Braidwood UFSAR, will continue to be in compliance with the Byron and Braidwood design basis acceptance criteria at the proposed MUR power uprate conditions. Also, confirm that the operating temperatures for support elements, as defined in Table 3.9-17 of the Byron and Braidwood UFSAR, are not affected by the MUR power uprate.

Response

The NSSS component supports, which include the reactor vessel, steam generator, reactor coolant pump, and pressurizer equipment supports, were assessed for the MUR power uprate as discussed in the response to EMCB R-7 and were shown to remain acceptable and bounded by the current design basis. Therefore, the NSSS component supports will remain in compliance with UFSAR Section 3.9.3.4.

The operating temperatures of the supports, as outlined in Table 3.9-17 of the UFSAR, are not affected by the MUR power uprate. The MUR power uprate does not require an increase in the ambient containment temperature design value. Further, the small changes to the NSSS design temperatures, as discussed in the response to EMCB R-7, do not require a change to the operating temperature of the supports attached to the steam generator, reactor coolant pump, reactor vessel, or pressurizer.

NRC/EMCB Request 12

Section IV.1.A.vi.1.b of Attachment 7 to the LAR discusses the structural evaluation of Byron and Braidwood Unit 1 replacement steam generators and states that a reconciliation analysis was performed to address the structural integrity of the entire steam generator pressure boundary for the MUR power uprate conditions. Discuss further information relative to, before and after uprate, the maximum stress intensity and the cumulative fatigue usage factors for the critical components of the primary and secondary sides, including nozzles, of the replacement steam generators and the respective service conditions. Also, confirm that the reconciliation analysis was performed in accordance with the original design code of record and in compliance with the Byron and Braidwood stations design basis acceptance criteria.

Response

During the structural integrity analysis of the replacement steam generators (RSGs) on Unit 1 for MUR conditions it was concluded that the maximum primary and secondary side temperatures and pressures specified for MUR power uprate conditions were less than the primary and secondary side temperatures and pressures specified for the original analysis. Therefore, there are no changes to the calculated stress values or limits for design conditions (i.e., name plate conditions).

However, a reconciliation analysis was performed for critical components of the replacement steam generators due to differences in the Level A & B (Normal and Upset), Level C (Emergency) and Level D (Faulted) condition loads. The stress intensities and cumulative usage factors for these service conditions for pre-MUR and post-MUR power uprate conditions are included in Tables EMCB R12-1 through R12-4.

The reconciliation analysis was performed in accordance with the original design code of record as required by the current Certified Design Specification. Specifically, the acceptance criteria

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for the reconciliation of the pressure boundary components were those specified in the 1986 ASME B&PV Code with no Addenda, for Section III, Class 1 components. The Code acceptance criteria are unchanged from the original RSG analysis.

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Table EMCB R12-1: Stress Intensity (SI) and Fatigue Usage Factors (FUF) for Level A & B Conditions							
Component / Location	MUR SI Range (ksi)	Orig. SI Range (ksi)	MUR SI Limit (ksi)	Orig. SI Limit (ksi)	MUR FUF	Orig. FUF	FUF Limit
Tubesheet							
Primary Head / Tubesheet Juncture	38.5*	82.1	80.1	87.3	0.880	0.741	1.0
Secondary Shell / Tubesheet Juncture	86.4	85.4	95.0	87.3	0.160	0.223	1.0
Tubesheet Perforated Region	90.1	90.0	95.0	93.6	0.330	0.387	1.0
Primary Nozzle							
Primary nozzle	67.85	67.85	80.1	80.1	0.839	0.839	1.0
Primary nozzle safe end	57.37	57.37	60.3	60.3	0.096	0.096	1.0
Primary Manway							
Cover	30.3	30.3	80.1	80.1	0.006	0.006	1.0
Shell/flange	46	46.0	80.1	80.1	0.121	0.121	1.0
Stud	See Table EMCB R12-4 for Average and Maximum Bolt Stresses				0.871	0.871	1.0
Primary Head Support Pad	79.4	79.4	80	80	0.67	0.67	1.0
Primary Divider Plate	63.9	63.9	69.9	69.9	0.905	0.904	1.0
Small Nozzles							
¾" Nozzles	13.96	11.83	26.7	26.7	0.81	0.679	1.0
Steam Drum/Cone/Lower Shell Assembly	74.22	62.9	80.1	80.1	0.025	0.021	1.0

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Table EMCB R12-1: Stress Intensity (SI) and Fatigue Usage Factors (FUF) for Level A & B Conditions							
Component / Location	MUR SI Range (ksi)	Orig. SI Range (ksi)	MUR SI Limit (ksi)	Orig. SI Limit (ksi)	MUR FUF	Orig. FUF	FUF Limit
8" Shell Cone Handhole							
Shell/cover/flange	67.3	57	80	80	0.256	0.074	1.0
Stud	See Table EMCB R12-4 for Average and Maximum Bolt Stresses				0.987	0.975	1.0
6" Feeding Handhole							
Shell/cover/flange	78.0	76.5	80	80	0.823	0.374	1.0
Stud	See Table EMCB R12-4 for Average and Maximum Bolt Stresses				0.823	0.84	1.0
2" Inspection Port							
Shell/cover/flange	77.6	65.8	80	80	0.214	0.205	1.0
Stud	See Table EMCB R12-4 for Average and Maximum Bolt Stresses				0.864	0.807	1.0
Secondary Manway							
Flange/Steam Drum Head	55.2	46.8	80	80	0.02	0.019	1.0
Diaphragm	60.4	60.4	69.9	69.9	0.02	0.015	1.0
Cover	25.5	21.6	80	80	0.02	0.000	1.0
Stud	See Table EMCB R12-4 for Average and Maximum Bolt Stresses				0.973	0.752	1.0

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Table EMCB R12-1: Stress Intensity (SI) and Fatigue Usage Factors (FUF) for Level A & B Conditions							
Component / Location	MUR SI Range (ksi)	Orig. SI Range (ksi)	MUR SI Limit (ksi)	Orig. SI Limit (ksi)	MUR FUF	Orig. FUF	FUF Limit
Pressure Boundary Attachments							
Seal Skirt Transition Juncture	44.6*	44.6*	56.1	56.1	0.538	0.476	1.0
Skirt Weld	41.2*	41.2*	56.1	56.1	0.74	0.559	1.0
Steam Drum Head/Steam Drum Juncture	48.6	48.6	80.1	80.1	0.401	0.209	1.0
Steam Drum / Trunion Juncture	64.8*	80	80.0	80.0	0.688	0.239	1.0
Primary Deck Lug/Steam Drum Juncture	72	61	80	80	0.608	0.546	1.0
Shroud Lug	40.6	34.4	58.5	58.5	0.652	0.545	1.0
Shroud Lug/ Shell Juncture	54.7	46.4	80	80.1	0.652	0.545	1.0
Upper Vessel Support/ Steam Drum Juncture	70.6	59.8	80.1	80.1	0.021	0.010	1.0
Main Feedwater Nozzle							
Shell/nozzle juncture	77.6	77.6	80	80	0.408	0.346	1.0
Nozzle	69.3	58.7	80	80	0.046	0.039	1.0
Transition ring/Thermal sleeve	27.2*	27.2*	69.9	69.9	0.985	0.945	1.0
Steam Outlet Nozzle							
Nozzle/Safe End Juncture	26.8	22.7	70	70	0	0	1.0
Nozzle	69.5	58.9	80	80	0.048	0.035	1.0
Steam Drum Head	71.3	60.4	80	80	0.049	0.033	1.0
Perforated Zone	76.7	65	80	80	0.080	0.059	1.0

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Table EMCB R12-1: Stress Intensity (SI) and Fatigue Usage Factors (FUF) for Level A & B Conditions							
Component / Location	MUR SI Range (ksi)	Orig. SI Range (ksi)	MUR SI Limit (ksi)	Orig. SI Limit (ksi)	MUR FUF	Orig. FUF	FUF Limit
Small Nozzles							
3" Blowdown Nozzle	12.02	10.19	26.7	26.7	0.85	0.928	1.0
3" Recirculation Nozzle	12.02	15	26.7	26.7	0.5	0.938	1.0
¾" Nozzles	13.96	11.83	26.7	26.7	0.81	0.679	1.0
Acoustic Sensor Pad	54.63	46.3	56	56	0.81	0.777	1.0
Tubes	73.8	73.8	79.8	79.8	0.19	0.19	1.0

* ***Bold/italicized*** stress range values were determined using simplified elastic-plastic analysis in accordance with NB-3228.5.

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Table EMCB R12-2 - Primary Membrane and Bending Stresses for Level C Conditions								
Component / Location	MUR Pm/PL SI (ksi)	Orig. Pm/PL SI (ksi)	MUR Pm SI Limit (ksi)	Orig. Pm SI Limit (ksi)	MUR PL/Pm+Pb SI (ksi)	Orig. PL/Pm+Pb SI (ksi)	MUR PL Pm+Pb SI Limit (ksi)	Orig. PL Pm+Pb SI Limit (ksi)
Primary Head / Tubesheet / Secondary shell	29.6	29.2	38.79	38.79	49.9	49.2	64.65	64.65
Primary Nozzle	Bounded by design conditions							
Primary Manway								
Cover	13.33	13.33	38.8	38.8	24.39	24.39	58.2	58.2
Shell/flange	21.31	21.31	38.8	38.8	21.31	21.31	58.2	58.2
Primary Head Support Pad	Bounded by design conditions							
Primary Divider Plate	Bounded by design conditions							
Small Nozzles	Bounded by design conditions							
Steam Drum/Cone/Lower Shell Assembly	Bounded by design conditions							
8" Shell Cone Handhole	29.3	29.02	29.37	29.37	32.6	32.2	48.06	48.06
6" Feedring Handhole	29.3	29.02	29.37	29.37	34.6	34.6	48.06	48.06
2" Inspection Port	10.6	10.5	28	28	20.7	20.5	42	42
Secondary Manway	Bounded by design conditions							

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Table EMCB R12-2 - Primary Membrane and Bending Stresses for Level C Conditions								
Component / Location	MUR Pm/PL SI (ksi)	Orig. Pm/PL SI (ksi)	MUR Pm SI Limit (ksi)	Orig. Pm SI Limit (ksi)	MUR PL/Pm+Pb SI (ksi)	Orig.PL/Pm+Pb SI (ksi)	MUR PL Pm+Pb SI Limit (ksi)	Orig. PL Pm+Pb SI Limit (ksi)
Pressure Boundary Attachments								
Seal Skirt Transition Juncture	Bounded by design conditions							
Skirt Weld	Bounded by design conditions							
Steam Drum Head/Steam Drum Juncture	Bounded by design conditions							
Steam Drum / Trunion Juncture	28.8	28.5	39.4	39.4	36	35.6	65.7	65.7
Primary Deck Lug/Steam Drum Juncture	29.8	29.8	43.8	43.8	65.2	65.2	65.7	65.7
Shroud Lug	2.32	2.3	26.37	26.37	5.8	5.73	43.95	43.95
Shroud Lug/ Shell Juncture	24.9	24.63	36.9	36.9	26.9	26.61	65.7	65.7
Upper Vessel Support/ Steam Drum Juncture	Bounded by design conditions							
Main Feedwater Nozzle								
Shell/nozzle juncture	29	28.7	43.8	43.8	46.6	46.1	65.7	65.7
Nozzle	28.6	28.3	43.8	43.8	28.6	28.3	65.7	65.7
Transition ring/Thermal sleeve	9.5	9.4	28	28	26.1	25.8	41.9	41.9
Steam Outlet Nozzle	Bounded by design conditions							
Small Nozzles	Bounded by design conditions							
Tubes	22.95	22.7	35.2	35.2	32.35	32	52.9	52.9
Tubes (external pressure)	0.168	0.166	1.424	1.424	--	--	--	--

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Table EMCB R12-3: Primary Membrane and Bending Stresses for Level D Conditions								
Component / Location	MUR Pm/PL SI (ksi)	Orig. Pm/PL SI (ksi)	MUR Pm SI Limit (ksi)	Orig. Pm SI Limit (ksi)	MUR PL/Pm+Pb SI (ksi)	Orig.PL/Pm+Pb SI (ksi)	MUR PL Pm+Pb SI Limit (ksi)	Orig. PL Pm+Pb SI Limit (ksi)
Primary Head / Tubesheet / Secondary Shell	29.6	29.1	56	56	68.7	67.8	84	84
Primary Nozzle								
Primary nozzle	51.51	51.51	56	56	76.27	76.27	84	84
Primary nozzle safe end	27.7	27.7	48.9	48.9	39.93	39.93	72.36	72.36
Primary Manway								
Cover	13.33	13.33	56	56	24.39	24.39	84	84
Shell/flange	21.31	21.31	56	56	21.31	21.31	84	84
Primary Head Support Pad	15.9	15.9	56	56	53.9	53.9	84	84
Primary Divider Plate	35.9	35.4	52.5	52.5	61.2**	60.4	67.5	67.5
Small Nozzles								
¾" Nozzles	16.9	16.7	42.8	42.8	38	37.5	64	64
Steam Drum/Cone/Lower Shell Assembly	46.2	35.6	56	56	61.5	60.7	84	84
8" Shell Cone Handhole	40.9	40.4	56	56	40.9	40.4	84	84
6" Feeding Handhole	35.3	34.8	56	56	35.3	34.8	84	84
2" Inspection Port	55.1	54.4	56	56	60.9	60.1	84	84
Secondary Manway	47.7	47.1	56	56	47.7	47.1	84	84

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Table EMCB R12-3: Primary Membrane and Bending Stresses for Level D Conditions								
Component / Location	MUR Pm/PL SI (ksi)	Orig. Pm/PL SI (ksi)	MUR Pm SI Limit (ksi)	Orig. Pm SI Limit (ksi)	MUR PL/Pm+Pb SI (ksi)	Orig.PL/Pm+Pb SI (ksi)	MUR PL Pm+Pb SI Limit (ksi)	Orig. PL Pm+Pb SI Limit (ksi)
Pressure Boundary Attachments								
Seal Skirt Transition Juncture	8.81	8.7	49	49	26.8	26.5	73.5	73.5
Steam Drum Head/Steam Drum Juncture	35.35	34.9	56	56	46.1	45.5	84	84
Steam Drum / Trunnion Juncture	35.8	35.3	56	56	42.7	42.2	84	84
Primary Deck Lug/Steam Drum Juncture	40.1	40.1	56	56	74	74	84	84
Shroud Lug	39	38.5	49	49	43.4	42.9	73.5	73.5
Shroud Lug/ Shell Juncture	33.5	33.1	56	56	39.5	39.1	84	84
Upper Vessel Support/ Steam Drum Juncture	28	27.6	56	56	63.9	63.1	84	84
Main Feedwater Nozzle								
Shell/nozzle juncture	33.9	33.5	56	56	83.8	83.3	84	84
Nozzle	7.9	7.8	56	56	29.5	29.1	84	84
Transition ring/Thermal sleeve	12.9	12.7	49	49	53	52.3	73.5	73.5
Steam Outlet Nozzle								
Pipe extension	16.68	16.47	42	42	43.38	42.82	63	63
Nozzle/Safe End Juncture	14.99	14.8	49	49	40.84	40.32	73.5	73.5
Nozzle	25.43	25.1	56	56	54.82	54.12	84	84
Steam Drum Head	26.34	26	56	56	55.84	55.12	84	84
Perforated Zone	32.62	32.2	56	56	61	60.22	84	84

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Table EMCB R12-3: Primary Membrane and Bending Stresses for Level D Conditions								
Component / Location	MUR Pm/PL SI (ksi)	Orig. Pm/PL SI (ksi)	MUR Pm SI Limit (ksi)	Orig. Pm SI Limit (ksi)	MUR PL/Pm+ Pb SI (ksi)	Orig.PL/ Pm+Pb SI (ksi)	MUR PL Pm+Pb SI Limit (ksi)	Orig. PL Pm+Pb SI Limit (ksi)
Small Nozzles								
3" Blowdown Nozzle	21.1	30.6	42.8	42.8	42.1	61.2	64	64
3" Recirculation Nozzle	21.1	30.6	42.8	42.8	42.1	61.2	64	64
Acoustic Sensor Pad	Bounded by Steam Drum/Cone/Lower Shell Assembly							
Tubes	31.4	31	56	56	68.68	67.8	84	84
Tubes (external pressure)	1.142	1.127	1.780	1.780	--	--	--	--

** A prorating factor corresponding to the SG secondary side level D loading has been applied to the Divider Plate MUR PL/Pm+Pb SI, making the reported value conservative. However, only primary stresses from divider plate level D loads need to be analyzed and since the primary side pressures are invariant between MUR and Original conditions, both the level D stresses and their ASME Code limits are unchanged.

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Table EMCB R12-4: Average and Maximum Stresses for Studs/Bolts

Component / Location	MUR Average Stress (ksi)	Orig. Average Stress (ksi)	MUR Average Stress Limit (ksi)	Orig. Average Stress Limit (ksi)	MUR Maximum Stress (ksi)	Orig. Maximum Stress (ksi)	MUR PL Maximum Stress Limit (ksi)	Orig. Maximum Stress Limit (ksi)
<u>Primary Manway</u>								
Level A/B	43.8	43.8	54.6	54.6	55.4	55.4	81.9	81.9
Level C	34.7	34.7	52.6	52.6	76	76	78.9	78.9
Level D	34.7	34.7	87.5	87.5	76	76	125	125
<u>8" Shell Cone Handhole</u>								
Level A/B	13.5	13.25	57.7	57.7	49.7	48.7	77.9	77.9
Level C	41	40.54	57.7	57.7	78.6	77.73	86.7	86.7
Level D	41.1	40.54	86.2	86.2	79.7	78.66	125	125
<u>6" Feeding Handhole</u>								
Level A/B	41.4	40.6	57.7	57.7	69	67.6	77.9	77.9
Level C	36.7	36.3	57.7	57.7	55.2	54.6	86.7	86.7
Level D	36.8	36.3	86.2	86.2	55.6	54.9	125	125
<u>2" Inspection Ports</u>								
Level A/B	40.9	40.1	57.7	57.7	52.8	51.8	77.9	77.9
Level C	41.1	40.6	57.7	57.7	47.4	46.9	86.7	86.7
Level D	41	40.5	86.2	86.2	47.1	46.5	125	125

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Table EMCB R12-4: Average and Maximum Stresses for Studs/Bolts

Component / Location	MUR Average Stress (ksi)	Orig. Average Stress (ksi)	MUR Average Stress Limit (ksi)	Orig. Average Stress Limit (ksi)	MUR Maximum Stress (ksi)	Orig. Maximum Stress (ksi)	MUR PL Maximum Stress Limit (ksi)	Orig. Maximum Stress Limit (ksi)
<u>Secondary Manway</u>								
Level A/B	47.8	40.5	57.7	57.7	72.1	61.1	77.9	77.9
Level C	32.1	31.8	57.7	57.7	58.5	57.9	77.9	77.9
Level D	30.4	30.0	86.2	86.2	44.7	44.1	125	125

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NRC/EMCB Request 13

Discuss further information to demonstrate that, for the expected post-uprate conditions, the spent fuel pool (SFP) structure, including SFP liner and the spent fuel racks, remain capable of performing their intended design functions and will continue to be in compliance with the Byron and Braidwood design basis code of record(s) and acceptance criteria.

Response

During a February 1, 2012 clarification call between Exelon Generation Company (EGC) and the Nuclear Regulatory Commission (NRC) staff, EGC requested and the NRC staff agreed to allow EGC to provide a response to this request under a separate transmittal at a later date.

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NRC Balance of Plant (NRC/SBPB)

NRC/SBPB Request 1

Technical Specification (TS) 3.7.4 for the steam generator (SG) power operated relief valves (PORVs) currently allows 24 hours completion time to restore all but one of the four PORVs when two or more PORVs are inoperable. Hence, the TS action statement would allow all four PORVs to be inoperable for up to 24 hours.

The analysis for a steam generator tube rupture (SGTR) credits the use of two PORVs to cool down the reactor coolant system (RCS) rapidly to achieve a subcooling margin in order to start depressurizing the RCS to stop the break flow. The analysis identifies the most limiting single failure as a failure of a SG PORV on an intact SG. Thus, the licensee credits the SG PORVs with a high significance for successfully mitigating a SGTR. The current TS that allows 24 hours for all four PORVs to be inoperable (loss of function) may not be appropriate.

Justify the current TS action statement that allows all four SG PORVs to be inoperable based on the new SGTR analysis.

Response

Based on discussions during the February 1, 2012 clarification call between EGC and the NRC staff, the NRC staff revised this request in an e-mail dated February 8, 2012 (Reference 4). The NRC staff agreed to allow EGC to provide a response to this request under separate transmittal at a later date.

NRC/SBPB Request 2

The licensee identifies the SG PORVs as being a key component in mitigating an SGTR from an overfill condition. The licensee identified an SG PORV failing to open on one of the intact SGs as the most limiting failure for the margin to overfill (MTO) analysis. The installation of an uninterruptible power supply was made to reduce the current vulnerability of a single failure making two SG PORVs inoperable.

In Table I-2, "Steam Generator Tube Rupture Equipment List," the licensee states, "Table I-2 identifies the systems, components, and instrumentation which are credited for accident mitigation." The Table I-2 does not list the SG PORV controllers.

Provide a description of the PORVs electrical systems to include power supplies to the controllers and circuitry, and include any other circuits that would affect the SG PORV's ability to perform its function; identify any shared components (i.e., electrical, mechanical, Instrumentation & Control, etc.); and justify not including the SG PORV controllers.

Response

As described in Technical Specifications Bases 3.7.4, a Steam Generator (SG) Power Operated Relief Valve (PORV) is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand. The definition of OPERABLE requires that all necessary attendant instrumentation, controls, normal or emergency electrical power required to perform its specified safety function are also capable of performing their related support functions. As such, the SG PORVs were listed as an assembly

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rather than listing individual components required to support the performance of their safety function.

The SG PORVs do not share mechanical components. The SG PORVs on a single electrical division share their normal source of 480 VAC from an Engineered Safety Feature (ESF) switchgear on that division. On Unit 1, for example, Division 1 Motor Control Centers (MCCs) are supplied from ESF Switchgear 131X and Division 2 MCCs are supplied from ESF Switchgear 132X. The SG PORVs on a single division share a common process control cabinet. On Unit 1, for example, SG PORVs 1A and 1D receive process control signals from cabinet 1PA33J and SG PORVs 1B and 1C receive process control signals from cabinets 1PA34J.

The existing SG PORVs are fed from safety related 480V MCCs which feed a power transformer in the 1/2MS018JA, JB, JC, and JD SG PORV control panels (controllers). The SG PORV controllers contain a 4KVA power transformer that reduces the 480VAC supply to a 125VAC control power source and a secondary AC power supply source that is subsequently rectified to a DC source and used to power the reversible hydraulic pump motor contained on the PORV operator. The SG PORV controllers receive a control signal from the pressure control loops associated with the steam line pressure controls from control cabinets 1/2PA33J and 1/2PA34J. This signal is generated based on pressure control or demands from a Manual Auto (MA) Station on the Main Control Board. The output signal from the controllers drives the hydraulic pump motor to either open or close the valve.

Once installed the modified SG PORVs will incorporate a battery-backup Uninterruptible Power Supply (UPS) into the power feed to one of two SG PORV circuits per electrical division (SG PORV's 1/2D for Division 1 and 1/2C for Division 2). The UPS will provide battery back-up power to the valve in the event of a loss of the UPS normal AC power supply.

The SG PORV UPS modification also affects the power source to the Division 1 SG PORV process control cabinets. A loss of power to a safety related Division 1 would also result in the loss of power to the 1/2PA33J control cabinet since the cabinets are fed from two separate 120VAC distribution panels from Division 1 MCCs. The pressure modulating signals for SG PORVs 1/2A and 1/2D are processed in 1/2PA33J. To resolve this issue, the 120VAC distribution panel feed to the 1/2PA33J cabinet is replaced with a feed from a Division 1 inverter backed instrument bus. The 1B and 1C SG PORV control circuits are processed in the 1/2PA34J panels which are currently fed from a Division 2 120VAC distribution panel and a Division 2 inverter backed instrument bus and thus would be unaffected by a Division 2 bus outage/failure.

Tables SBPB R2-1 and R2-2 below list the power supplies for the SG PORV controllers and associated control cabinets. The information provided reflects the configuration following the implementation of the UPS modification. All power supplies are Electrical Class 1E.

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Table SPBP R2-1: SG PORV Power Supplies			
SG PORV	Control Panel (Controllers)	Primary Power Supply	Backup Power Supply
Byron Station			
1MS018A	1MS018JA	131X2B	
1MS018B	1MS018JB	132X1	
1MS018C	1MS018JC	UPS from 132X5	UPS from Battery
1MS018D	1MS018JD	UPS from 131X4	UPS from Battery
2MS018A	2MS018JA	231X2B	
2MS018B	2MS018JB	232X1	
2MS018C	2MS018JC	UPS from 232X5	UPS from Battery
2MS018D	2MS018JD	UPS from 231X4	UPS from Battery
Braidwood Station			
1MS018A	1MS018JA	131X2B	
1MS018B	1MS018JB	132X1	
1MS018C	1MS018JC	UPS from 132X5	UPS from Battery
1MS018D	1MS018JD	UPS from 131X4	UPS from Battery
2MS018A	2MS018JA	231X2B	
2MS018B	2MS018JB	232X1	
2MS018C	2MS018JC	UPS from 232X5	UPS from Battery
2MS018D	2MS018JD	UPS from 231X4	UPS from Battery

Table SBPB R2-2: Control Cabinet Power Supplies		
Control Cabinet	Primary Power Supply	Backup Power Supply (120 VAC Distribution Panel)
Byron Station		
1PA33J	Instrument Bus 113	131X1
1PA34J	Instrument Bus 114	132X1
2PA33J	Instrument Bus 213	231X1
2PA34J	Instrument Bus 214	232X1
Braidwood Station		
1PA33J	Instrument Bus 113	131X1
1PA34J	Instrument Bus 114	132X1
2PA33J	Instrument Bus 213	231X1
2PA34J	Instrument Bus 214	232X1

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The licensee is making modifications to the auxiliary feedwater (AFW) flow control valves to include an air accumulator tank capable of supplying air for 30 minutes. In accordance with their analysis, AFW flow control is required longer than 30 minutes to mitigate the SGTR and for RCS cool down. In Attachment 5a, Section II.2.E, Single Failure Considerations, the licensee states:

In addition, since the failure of an intact SG PORV scenario assumes a loss of offsite power with an associated loss of Instrument Air (IA), the modification described in Section II.2.F, Item 1, assures that AFW flow control is maintained throughout the event.

According to the licensee's evaluation, an SGTR event continues until break flow is terminated at 3458/3258 seconds (Units 1 and 2).

Describe the basis for selecting 30 minutes, and explain how the amount of air that is required is determined and the amount of air available to support this function.

Response

As noted in the NRC's request above, the limiting Steam Generator Tube Rupture (SGTR) event continues until break flow is terminated (3,458 seconds Unit 1 and 3,258 seconds for Unit 2). Attachment 5a to the MUR power uprate LAR (Reference 1), Section II.2.F, "Modifications to Support MTO Single Failure Considerations," describes the plant modifications Byron and Braidwood Stations will be implementing to support the Steam Generator Margin to Overfill Reanalysis assumptions. Included in these modifications will be the installation of two instrument air accumulator tanks on each Unit (one per train) to provide a safety related air supply for the Auxiliary Feedwater (AFW) Flow Control Valves (FCVs)(AF005). The air accumulator tanks for the AFW FCVs (AF005) are only required for the first 30 minutes (1,800 seconds) post-SGTR event initiation for AFW flow control and isolation. After AFW flow is isolated to the ruptured SG, AFW flow control to the ruptured SG is no longer needed for the duration of the event. AFW flow to the non-ruptured SGs is controlled by throttling either the AFW FCVs (AF005) or the motor operated AFW valves (AF013); these valves are in series with each other.

The Emergency Operating Procedures (EOPs) (1/2B(w)EP-0, Reactor Trip or Safety Injection Unit 1(2)) direct isolation of AFW to the ruptured SG with the motor operated AFW isolation valves (AF013). Following the installation of the air accumulator tanks, the EOPs will be revised to direct the closure of the AFW FCVs (AF005) via the controller in the Main Control Room at the same point in the procedure that they are directed to close the AFW (AF013) valve. If an AF0013 valve fails to close, then the EOPs will direct an operator to be dispatched to close the associated AF005 flow control valve locally. This action prevents the valve from failing open when the air supply from the accumulator tank is exhausted. It was determined that a 30 minute supply of air is sufficient to allow the operator to reach the AF005 valve and manually close it using the installed handwheel on the valve.

The time assumed for the local closure of the AF005 valves is consistent with the current Byron and Braidwood design basis. Specifically, in UFSAR Sections 3.11.10, "High Energy Line Break (HELB)," 10.4.9.3, "Auxiliary Feedwater, Safety Evaluation," and 15.2.8.2, "Feedwater System Pipe Break, Analysis of Effects and Consequences," for feedline and main steamline breaks,

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operator action is credited to isolate auxiliary feedwater to the faulted steam generator within 20 minutes.

The AF005 instrument air accumulators were sized to include 30 minutes of air supply as described above and additional capacity to account for:

- Stroking four valves (1 Train) from full open to full closed,
- Maximum air consumption rate for four electric to pneumatic signal converters (IY's),
- Maximum air consumption rate for four valve positioners, and
- 10% allowance for leakage.

The total volume required was determined to be 27.3 cubic feet (204 gallons). Additional conservatism exists since the tank size is 33.4 cubic feet (250 gallons). This ensures that adequate air is available to support the required function of AFW flow control and isolation.

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Figure II-5 of Attachment 5a shows the SG water volume on Unit 1 trending towards the maximum available quantity. At approximately 3200 seconds, the trend tapers off, resulting in a margin to overfill of approximately 94 cubic feet. At the same time other graphs show a sharp reduction in SG pressure, which logically corresponds to a second opening of the SG PORVs on the intact SGs. This action stops the upward trend and prevents the overfill condition. The licensee does not identify a critical operator action to open the SG PORVs a second time within a certain time period as a condition to prevent an overfill of the SG.

In the updated final safety analysis report, Section 15.6.3.2, under the section describing major operator actions, the licensee's analysis credits operators for reopening pressurizer PORV, four minutes after establishing normal charging and letdown, in order to equalize the RCS and SG pressures.

In Attachment 5a (page II-10), the licensee states that the SG PORVs on the intact SGs automatically open, as necessary, to maintain RCS subcooling margin. The above mentioned graph trend shows a sharp pressure reduction at 3200 seconds, which is not indicative of SG PORV automatically controlling pressure at a prescribed setpoint.

- a. *Evaluate whether this operator action is credited to be performed within a specific time in order to prevent an overfill condition.*
- b. *If operator action is required, identify the action as a critical operator action.*
- c. *Describe whether the new analysis changes the existing UFSAR analysis, and results in the major operator action opening a SG PORV rather than a pressurizer PORV after SI termination to stop an overfill condition from occurring.*

Response

The Steam Generator Tube Rupture/Margin to Overfill (SGTR/MTO) analysis methodology used in the new SGTR/MTO Analysis submitted in Attachment 5a to the MUR power uprate LAR (Reference 1) is different from the methodology in the current Analysis of Record (AOR) described in the UFSAR Section 15.6.3, "Steam Generator Tube Rupture." The methodology used in the current AOR SGTR/MTO analysis explicitly models operator actions after Safety Injection (SI) flow termination (i.e. securing Emergency Core Cooling (ECCS) flow), including the operator action to open the pressurizer PORV within a specific time in order to prevent an overfill condition. The SGTR/MTO analysis provided in Attachment 5a, "Steam Generator Tube

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Rupture Analysis report,” of the MUR power uprate submittal (Reference 1) uses the NRC approved methodology described in WCAP-10698-P-A, “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill” (Reference 5).

Consistent with the WCAP-10698-P-A methodology, specific operator actions after SI termination are not used and the LOFTTR2 computer code is used to predict the transient responses that lead to pressure equalization (break flow termination) and to demonstrate the SG overfill condition is not reached. Therefore, actions taken after SI termination are not considered critical operator responses and as such are modeled to occur as conditions require as predicted by the LOFTTR2 computer code.

As discussed in Section II.2.D, “Operator Action Times,” of Attachment 5a of the MUR power uprate submittal (Reference 1), the critical operator responses are:

1. Isolate Auxiliary Feedwater (AFW) flow to the ruptured Steam Generator (SG),
2. Isolate the MSIV on the ruptured SG,
3. Initiate RCS cooldown, to initiate RCS depressurization, and
4. Terminate Safety Injection (SI) (secure Emergency Core Coolant (ECCS) flow).

These operator actions and the corresponding operator action times used for the analyses are summarized in Table II-2, “Operator Action Times for Design Basis SGTR Analyses” of Attachment 5a of the MUR power uprate submittal (Reference 1). These actions are consistent with the actions in WCAP-10698-P-A (Reference 5) Table 2.3-2, “Operator Action Times for Design Basis SGTR Analysis.” Also, consistent with the methodology in WCAP-10698-P-A (Reference 5) the times required for cooldown, depressurization, and pressure equalization are calculated using the LOFTTR2 program. The analyses do not model specific operator action times after SI termination.

In accordance with Emergency Operating Procedures (EOPs) (1/2B(w)EP-3), the same step that directs the operator to terminate RCS cooldown also directs the operators to maintain RCS temperature below the required temperature. This step occurs before SI termination and is a step that is monitored and acted on throughout the procedure. SI termination occurs at 2,311 seconds on Unit 1 and at 2,482 seconds on Unit 2. After SI termination, LOFTTR2 models the opening of two of the intact SG PORVs to maintain the required RCS temperature from the EOPs. This action is predicted by LOFTTR2 to occur at approximately 3,200 seconds (Unit 1 analysis). This modeling is consistent with the methodology in WCAP-10698-P-A.

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Calculation Westinghouse commercial atomic power (WCAP) -10698-P-A provides a general assessment of the MTO for Westinghouse type reactors. There were instances where the licensee deviated from the input parameters selected in WCAP-10698-P-A as the most conservative.

- a. Decay heat is one of the input factors that influence MTO analyses and Thermal/Hydraulic analyses during a tube rupture. For the MTO analysis, the licensee states that plant specific sensitivities were performed for Bryon and Braidwood Units 1 and 2. These studies concluded that the 1979-2 σ American Nuclear Society (ANS) decay heat factor was more conservative compared to the 1971 +20% ANS decay heat model specified in WCAP-10698-P-A.*

Justify use of the 1979-2 σ ANS decay heat factor was more conservative compared to the 1971 +20% ANS decay heat factor.

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- b. *Similar to above, in determining the most conservative input values, the licensee chose to model the minimum AFW enthalpy of 0.03 Btu/lbm; whereas, WCAP-10698-P-A models the maximum temperature of AFW (maximum enthalpy) as the most conservative parameter in the analysis for MTO.*

Justify how the use of the minimum AFW enthalpy is more conservative compared to using the maximum temperature (enthalpy) for AFW.

Response

WCAP-10698-P-A (Reference 5) identified high decay heat and high Auxiliary Feedwater (AFW) temperature to be the conservative assumptions for the steam generator tube rupture margin to overfill (MTO) analysis. NSAL-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology" (Reference 6), identified a lower decay heat can be more limiting for some plants. To resolve the concerns of NSAL-07-11, plant-specific sensitivities were performed for Byron and Braidwood Units 1 and 2 to justify the decay heat model and AFW enthalpy assumed in the analysis. The Tables SBPB R5-1 and 2 show the impact on the Margin to Overfill (MTO) resulting from the sensitivity study. The study covered the T_{avg} range and the steam generator tube plugging levels supported by the analysis provided in Attachment 5a. The impact on MTO provided is relative to the limiting case modeling the ANS 1979 - 2σ decay heat model, low AFW enthalpy, low T_{avg} , and high steam generator tube plugging level.

The results show that use of the ANS 1971 + 20% decay heat model (cases 1 to 4) clearly provides more MTO margin than the ANS 1979 - 2σ decay heat model (cases 5 to 8). The conservative direction for AFW enthalpy is studied using low decay heat. Comparing cases 5 to 8 with corresponding cases 9 to 12 show that minimum AFW enthalpy is conservative.

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Table SBPB R5-1: Byron/Braidwood Unit 1 Results of Sensitivity Study on MTO		
Case	Description	Impact on MTO* (ft³)
1	Low T _{avg} , 5% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+321
2	Low T _{avg} , 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+314
3	High T _{avg} , 5% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+458
4	High T _{avg} , 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+457
5	Low T _{avg} , 5% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+47
6	Low T _{avg} , 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+52
7	High T _{avg} , 5% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+178
8	High T _{avg} , 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+176
9	Low T _{avg} , 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	Limiting case
10	Low T _{avg} , 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+3
11	High T _{avg} , 5% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+125
12	High T _{avg} , 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+123

* + indicates increase in MTO from the Limiting Case.

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Case	Description	Impact on MTO* (ft³)
1	Low T _{avg} , 10% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+337
2	Low T _{avg} , 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+353
3	High T _{avg} , 10% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+440
4	High T _{avg} , 0% tube plugging, ANS 1971 + 20%, maximum AFW enthalpy	+472
5	Low T _{avg} , 10% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+67
6	Low T _{avg} , 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+102
7	High T _{avg} , 10% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+176
8	High T _{avg} , 0% tube plugging, ANS 1979 - 2σ, maximum AFW enthalpy	+212
9	Low T _{avg} , 10% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	Limiting case
10	Low T _{avg} , 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+23
11	High T _{avg} , 10% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+129
12	High T _{avg} , 0% tube plugging, ANS 1979 - 2σ, minimum AFW enthalpy	+159

* + indicates increase in MTO from the Limiting Case.

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REFERENCES

- 1 Letter from Craig Lambert (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated June 23, 2011
- 2 Letter from Kevin F. Borton (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated December 9, 2011
- 3 Letter from N. J. DiFrancesco (U. S. NRC) to M. J. Pacilio (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 and Byron Station, Unit Nos. 1 and 2 – Request for Additional Information RE: Measurement Uncertainty Power Uprate Request (TAC NOS. ME6587, ME6588, 6589, AND ME6590)," dated November 28, 2011
- 4 E-mail from Brenda Mozafari (U. S. NRC) to Leslie Holden, et. al., Exelon Generation Company), "FW: Draft Balance of Plant RAIs related to MUR dated June 23, 2011," dated February 8, 2012
- 5 WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987
- 6 NSAL-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," November 2007.