NRR-PMDAPEm Resource

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To: Leslie Holden
Subject: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 4
Attachments: MUR Package for Proprietary and Factual Error Review - Part 4.docx

Leslie,

Here is Part 4 of the MUR package for Proprietary and Factual Error Review.

Joel

Hearing Identifier: NRR_PMDA
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From: Wiebe, Joel

Created By: Joel.Wiebe@nrc.gov

Recipients:
"Leslie Holden" <Leslie.Holden@exeloncorp.com>
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3.4 Engineering and Materials

3.4.1 Reactor Vessel Integrity and Internals

The NRC staff's review in the area of reactor vessel (RV) integrity focuses on the impact of the proposed power on pressurized thermal shock (PTS) calculations, neutron fluence calculations, RV pressure-temperature (P-T) limits, upper shelf energy (USE) evaluations, and the RV surveillance capsule withdrawal schedules. This review was conducted, consistent with the guidance contained in RIS 2002-03, to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR Part 50, Sections 50.60 and 50.61, and 10 CFR Part 50, Appendices G and H, following implementation of the proposed power.

Pressurized Thermal Shock (PTS)

Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of the PWR RV beltline materials to failure during a PTS event to assure that adequate fracture toughness exists during reactor operation. The NRC staff's requirements, methods of evaluation, and safety criteria for PTS assessments are in 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The NRC staff's review covered the PTS methodology and the calculations for the reference temperature for PTS (RTPTS) at the expiration of the license, considering neutron embrittlement effects.

Technical Evaluation

Section IV.1.C.i of Attachment 5 to the June 23, 2011, submittal, stated that PTS calculations were performed for Braidwood and Byron units using the current 40-year end of license (EOL) neutron fluence values and all the Braidwood and Byron units RV materials will continue to meet the 10 CFR 50.61 PTS screening criteria.

The PTS screening criteria are 270 ºF for RV plates, forgings, and axial welds and 300 ºF for RV circumferential welds. For the Braidwood units, the limiting RTPTS values reported in Section IV.1.C.i of Attachment 5 are 98 ºF for both units' intermediate to lower shell forging circumferential weld. The staff found that the limiting RTPTS values in the Braidwood 1 PTLR, Revision 4, and Braidwood, Unit 2, PTLR, Revision 4, for the corresponding RV materials are 99 ºF for Unit 1 and 98 ºF for Unit 2. The discrepancies between these two sources are insignificant, and the staff determined that they have no impact on the Braidwood units PTS evaluations. For the Byron units, the limiting RTPTS values reported in Section IV.1.C.i of Attachment 5 are 109 ºF for the intermediate shell forging for Byron Unit 1 and 114 ºF for the intermediate to lower shell forging circumferential weld for Byron Unit 2.

However, the NRC staff found that the limiting RTPTS values in the Byron, Unit Nos. 1 and 2, PTLRs (2006) for the corresponding RV materials are 110 ºF for Unit 1 and 116 ºF for Unit 2. The discrepancies between these two sources are within 2 ºF, and the staff determined that they have no impact on the Byron units PTS evaluations because the highest number is below the lowest PTS screening criteria of 270 °F

Conclusion

Since the RTPTS values for the limiting RV beltline materials of Braidwood, Units 1 and 2, and Byron, Units 1 and 2, are lower than the PTS screening criterion of 270 °F for forging and 300 °F for circumferential welds, the NRC staff concludes that after implementation of the MUR PU, the Braidwood and Byron units RV beltline materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61 and maintain structural integrity during a PTS event. The NRC staff has determined that the changes identified in the proposed LAR will not significantly impact the remaining safety margin.

Pressure and Temperature (P-T) Limits and Upper Shelf Energy (USE)

Regulatory Evaluation

Appendix G of 10 CFR Part 50, provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the reactor coolant pressure boundary (RCPB), including requirements on the USE values used for assessing the safety margins of the RV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of the USE assessments covered the impact of the MUR PU on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of effective full-power years (EFPYs) specified for the P-T limits, considering neutron embrittlement effects on the RV beltline materials under the proposed uprate.

Technical Evaluation

The NRC staff found that the current P-T Limits and low temperature overpressurization protection system (LTOPS) setpoints in the 2006 Byron Unit 1 PTLR are based on one quarter or three quarters of the RV wall thickness (¼T or ¾T) adjusted reference temperature (ART) values of 106 °F and 97 °F for the limiting material – the intermediate shell forging. Byron Unit 2 has two limiting materials: the ¼T and ¾T ART values for the circumferential weld are 107 °F and 89 °F; the corresponding values for nozzle shell forging are 52 °F and 37 °F.

In its submittal the licensee stated that,

> [f]or Unit1, the limiting ART values used in the development of the current P-T limit curves at 32 EFPY [effective full-power year] bound the …[uprate limit ART values (at 32 EFPY).

The NRC staff accepts this conclusion because the maximum fluence value on record (i.e., the 2006 Byron 1 PTLR) bounds the MUR maximum value reported in Table IV.1.C.ii-1 of Attachment 5. For Byron Unit No. 2, the licensee stated in the application,

> [f]or Unit 2, the limiting ART values used in the development of the current P-T limit curves at 32 EFPY are slightly lower than the …[uprate] limiting ART values (at 32 EFPY).

The NRC staff was unable to verify this statement because the maximum fluence value on record (i.e., the 2006 Byron Unit 2 PTLR) also bounds the MUR maximum value reported in Table IV.1.C.ii-1. In the November 1, 2011, supplement, the licensee stated that the uprate

neutron fluence value calculated specifically for the Byron Unit 2 nozzle shell forging at 32 EFPY is greater than the neutron fluence value used in the development of the current P-T limit curves in the 2006 Byron 2 PTLR for the nozzle shell forging at 32 EFPY, and this neutron fluence increase for the Byron 2 nozzle shell forging material resulted in higher ART values at 32 EFPY for the uprate as compared to those used in the development of the current P-T limit curves. The response further stated that the PTLR will be updated to reflect the uprate ART values and the 30.5 EFPY specified for the Byron 2 P-T limit curves. Since the PTLR will be updated to reflect the uprate ART values and the revised 30.5 EFPY for the Byron 2 P-T limit curves, the NRC staff determined that this issue is resolved and the licesnee's evaluation is acceptable.

For the Braidwood, Units 1 and 2, the licensee stated that the current P-T limits in Braidwood, Unit 1, PTLR, Revision 4, and Braidwood, Unit 2, PTLR, Revision 4, are licensed through 32 EFPY. It further stated that, "[t]he limiting ART values used in the development of the current P-T limit curves at 32 EFPY bound the MUR [PU] limiting ART values (at 32 EFPY) for both Units. Therefore, the current heatup and cooldown curves are valid through EOL (32 EFPY) with the MUR [PU] and do not require update."

The NRC staff accepts this conclusion because the maximum fluence value on record (i.e., the Braidwood 1 PTLR, Revision 4, and the Braidwood 2 PTLR, Revision 4), bounds the MUR PU maximum value reported in Table IV.1.C.ii-1 of Attachment 5. This means that the limiting ART remains unchanged considering MUR PU. Therefore, the current Braidwood, Units 1 and 2, P-T limits and LTOPS setpoints based on the same limiting ARTs remain valid through 32 EFPY.

For the USE evaluation, Section IV.1.C.v of Attachment 5 to the June 23, 2011, submittal, stated that the projected EOL Charpy USE decreases due to MUR PU fluence at the ¼T location were calculated per RG 1.99, Revision 2. It further stated that the limiting projected ¼T USE values are 75 ft-lbs for the intermediate to lower shell forging circumferential weld for Braidwood 1, 66 ft-lbs for the intermediate to lower shell forging circumferential weld for Braidwood, Unit 2, 65 ft-lbs for the nozzle to intermediate shell forging circumferential weld for Byron, Unit No. 1, and 68 ft-lbs for the nozzle to intermediate shell forging circumferential weld for Byron, Unit No. 2. However, the calculation details of these limiting USE values or their reference is not given in Attachment 5. The 2006 Byron PTLRs and the Braidwood PTLTs, Revision 4, contain no USE estimates either. By letter dated October 12, 2011, the NRC staff requested additional information.

In its November 1, 2011, response, the licensee provided the information requested, including detailed calculation of EOL USEs for the beltline materials of Braidwood and Byron units, supporting the MUR UP request. The staff performed independent calculations and found that minor discrepancies exist between the provided EOL USEs and the staff's values, primarily due to use of different initial USEs. The initial USEs used by the staff are based on the NRC's reactor vessel integrity database (RVID) which was updated in 2001 to reflect the NRC staff's review of the power increase request for the Braidwood and Byron units as documented in the approved Amendment No. 113 for Braidwood, Units 1 and 2, and approved Amendment No. 119 for Byron, Unit Nos. 1 and 2 (dated May 4, 2001). Using the staff's initial USE values for the limiting materials mentioned above (75 ft-lbs for all four limiting materials) instead of the licensee's values (80 ftlbs for Braidwood, Units 1 and 2, and Byron, Unit No. 2, and 77 ft-lbs for Byron, Unit No. 1), the NRC staff's arrived at EOL USEs of 70.3 ft-lbs for the limiting material of Braidwood, Unit 1, 61.9 ft-lbs for Braidwood, Unit 2, 63.3 ftlbs for Byron, Unit No. 1, and 63.8 ft-lbs for Byron, Unit No. 2. In summary, both the licensee's and the NRC staff's calculated EOL

USEs for the Braidwood and Byron units are above 50 ft-lbs as required by 10 CFR Part 50, Appendix G.

Conclusion

The licensee addressed the impact of the MUR PU on the Braidwood, Units 1 and 2, and Byron, Unit Nos. 1 and 2, USE evaluations. These analyses are documented in Attachment 5 to the licensee's letter of June 23, 2011, as supplemented by the November 1, 2011, response, to the NRC staff RAI. Since both the licensee's and the staff's calculated EOL USEs for the beltline materials of the Braidwood and Byron units are above 50 ft-lbs, the staff concludes that the RV beltline materials for the Braidwood and Byron units will continue to satisfy the EOL USE criteria specified in 10 CFR Part 50, Appendix G, at the proposed power.

The NRC staff has determined that the changes identified in the proposed LAR will not impact the remaining safety margin. Therefore, the NRC staff finds the proposed power to be acceptable with respect to the P-T limits and USE.

Reactor Vessel (RV) Material Surveillance Program

Regulatory Evaluation

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Appendix H of 10 CFR Part 50 provides the staff's requirements for the design and implementation of the RV material surveillance program.

Technical Evaluation

In its June 23, 2011, application, the licensee stated that the NRC-approved RV surveillance capsule withdrawal schedules for the Braidwood and Byron units are contained in the PTLR for each unit. It further stated that the current capsule withdrawal schedule in the PTLRs will be updated to reflect the latest capsule fluence, lead factor, and withdrawal EFPY associated with each capsule. The updated capsule withdrawal schedules for Byron, Unit Nos. 1 and 2, can be found in Tables IV.1.C.vi-1 and IV.1.C.vi-2, and for Braidwood, Units 1 and 2, in Tables IV.1.C.vi-3 and IV.1.C.vi-4 of Attachment 5 to the submittal. However, the source or reference for the latest capsule fluence, lead factor, and withdrawal EFPY associated with each capsule is not given. In its October 12, 2011, letter, the NRC staff requested additional information.

In its November 1, 2011, response, the licensee stated that the vessel and surveillance capsule fluence values contained in the submittal were calculated as part of the MUR PU project and are not contained in any prior surveillance capsule reports. Since the revised fluence calculations were based on the methodologies in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," and WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," which meet the requirements of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," the revised capsule fluence, lead factor, and withdrawal EFPY associated with each capsule and the RV fluence values are acceptable. This issue is resolved.

The surveillance program requirements in Appendix H of 10 CFR Part 50 were established to monitor the radiation-induced changes in the mechanical and impact properties of the RV

materials. Appendix H of 10 CFR Part 50 requires licensees to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region of light-water nuclear power reactors. Appendix H of 10 CFR Part 50 states that the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of American Standard Testing of Materials (ASTM) E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," that is current on the issue date of the ASME Code, to which the RV was purchased. Later editions of ASTM E 185 may be used including those editions through 1982 (i.e., ASTM E 185-82). This evaluation is limited to the current 40-year period of operation for these units.

Section IV.1.C.vi of Attachment 5 indicated that the surveillance capsule withdrawal schedules are consistent with ASTM E 185-82. Table 1 of ASTM E 185-82 requires that either a minimum of three, four, or five surveillance capsules be removed from each of the vessels, as based on the projected nil-ductility reference temperature shift (ΔRTNDT) of the limiting material at the cladvessel interface location of the RV at the EOL. Since Braidwood, Units 1 and 2, and Byron, Unit Nos. 1 and 2, PTLRs indicated that the EOL ΔRTNDT values of each unit's limiting materials are less than 100 ºF, the staff determined that each Braidwood or Byron unit only needs three surveillance capsules to meet the ASTM E 185-82 requirement. The RV materials surveillance program for each Braidwood or Byron unit contains six capsules, three are designated as required and three are standby. The three required capsules for each unit were withdrawn and tested to support current license operation. The staff compared the capsule withdrawal EFPYs in Table IV.1.C.vi-1 and Table IV.1.C.vi-2 for Byron, Unit Nos. 1 and 2, and TableIV.1.C.vi-3 and Table IV.1.C.vi-4 for Braidwood, Units 1 and 2, with those in the ASTM E 185-82 for plants with three capsules and concludes that the licensee's surveillance capsule withdrawal schedules in the LAR for the Braidwood and Byron units are in accordance with ASTM E 185-82 and are, therefore, acceptable. All standby capsules have also been withdrawn to avoid excessive irradiation and for future use, but have not been tested. Reinsertion of the withdrawn standby capsules is beyond the ASTM E 185-82 requirements for supporting current license operation.

Conclusion

The NRC staff concludes that the licensee's surveillance capsule withdrawal schedules in the LAR for Braidwood, Units 1 and 2, and Byron, Unit Nos. 1 and 2, are acceptable because all required capsules have already been withdrawn in accordance with the requirements of ASTM E 185-82 to support the current 40-year license operation. The revised capsule fluence, lead factor, and withdrawal EFPY associated with each capsule and the RV fluence values in the LAR are based on additional information provided in the licensee's supplement dated November 1, 2011, and are acceptable as discussed above.

3.4.2 RV Internals and Core Support Structures

Regulatory Evaluation

The RV internals and core support structures include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant pressure boundary). The NRC's acceptance criteria for RV internals and core support structures are based on GDC 1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RV internals and core supports. Matrix 1 of NRC Review Standard (RS)-001, Revision 0, "Review Standard for Extended Power

Uprates" (ADAMS Accession No. ML033640024), provides references to the NRC's approval of the recommended guidelines for RV internals in TRs WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (ADAMS Accession No. ML010430375) and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (ADAMS Accession No. ML003708443).

The SE dated December 16, 2011 (ADAMS Accession No. ML11308A770), on materials reliability program (MRP) report 1016596 (MRP-227), Revision 1, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines [I&E]," provides the NRC's evaluation of the industry's recommended I&E guidelines for RV internals which summarized the industry effort on this issue for the past few years. The NRC staff considers the MRP-227 report, as modified by the NRC staff, a replacement of the WCAP-14577, Revision 1-A, report and the BAW-2248-A report and will make necessary changes in the next revision of NRC RS-001.

Technical Evaluation

The licensee discussed the impact of the Braidwood, Units 1 and 2, and Byron, Unit Nos. 1 and 2, MUR PU on the structural integrity of the RV internals in Attachment 5 of the LAR, Section IV.1.A.ii and the licensee's November 1, 2011, supplement. The licensee concluded that the RV internals continue to meet their design criteria at proposed power conditions.

The staff reviewed the licensee's evaluation of the structural integrity of the Braidwood and Byron RV internals using NRC RS-001, Revision 0. Table Matrix-1 of NRC RS-001, Revision 0, provides the staff's gudiance for evaluating the potential for extended power uprates to induce aging effects on RV internals. Depending on the magnitude of the projected RV internals fluence, Table Matrix-1, may be applicable to the MUR application. However, the WCAP-14577, Revision 1-A, report and the BAW-2248-A report cited in Matrix 1 are no longer applicable since issuance of the SE on the MRP-227 report, which summarized most recent industry recommended I&E guidelines for PWR RV internals. Section IV.1.A.ii of Attachment 5 to the EGC's June 23, 2011, submittal, provides information for only a few RV internals and, therefore, appears incomplete. In its letter dated October 12, 2011, the staff requested additional information.

In its November 1, 2011, response, the licensee stated that the Exelon PWR Reactor Internals Management Program provides the necessary oversight and management to ensure that the integrity and operability of RV internals are consistent with the MRP-227 report requirements. The response further stated that (1) Exelon is an active participant in the MRP efforts relative to the development of the MRP-227 report for PWR RV internals inspections, (2) Exelon will prepare the necessary RV internals aging management plan in accordance with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report,"
Revision 2, as part of the overall license renewal process, of which Aging Management Program (AMP) XI.M16A provides recommended content for an acceptable PWR RV Internals AMP that references the MRP-227 report, and (3) an inspection plan will be submitted with the License Renewal Application, consistent with Category D plants as discussed in NRC RIS 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," dated July 21, 2011.

Since the licensee confirmed that Exelon has participated in the industry's initiatives on age-related degradation of all PWR RV internals, including core support structures and plans to submit its plant-specific program consistent with the MRP-227 report guidelines, the NRC staff concludes that the licensee's November 1, 2011, response is adequate to resolve the questions

in the NRC staff's October 12, 2011, letter. Exelon's participation in the industry's initiatives as described in its November 1, 2011, response also provides reasonable assurance that the Braidwood and Byron units' RV internals AMPs will be developed or modified from the current AMPs in accordance with the MRP-227-A report. In addition, the Braidwood and Byron units MUR PU results in very small changes to aging parameters such as temperature and neutron flux. Based on the above, the staff determined that the licensee's RV internals evaluation considering uprated conditions is acceptable.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the impact that uprate conditions will have on the structural integrity assessments for the RV internals. The staff has determined that the licensee's RV internals evaluation considering the uprate is acceptable because: (1) the Braidwood and Byron units' RV internals AMPs will be developed or modified from the current AMPs in accordance with the MRP-227-A report and (2) the Braidwood and Byron units uprate results in very small changes to aging parameters such as temperature and neutron flux. The staff has determined that the changes identified in the proposed request will not significantly impact the remaining safety margin.

3.4.3. Mechanical and Civil Engineering

3.4.3.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power, referred to as the CLTP. 10 CFR Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing ECCS analyses for LOCAs). This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in these analyses. Appendix K to 10 CFR Part 50 allows licensees to assume a power level less than 1.02 times the licensed power level (but not less than the licensed power level), provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties.

Section 3.1, "Conformance with NRC General Design Criteria," of the Byron and Braidwood UFSAR states that the design of both Braidwood and Byron fully satisfy and are in compliance with the intent of Appendix A to 10 CFR 50 "General Design Criteria for Nuclear Power Plants." The NRC staff's assessment of the proposed MUR PU in the areas of mechanical and civil engineering and the acceptance criteria are based on continued conformance with the requirements of the design and licensing basis of Braidwood and Byron.

The primary guidance used by Exelon and other licensees for LARs involving MUR PU is outlined in RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," which provides licensees with a guideline for organizing LAR submittals for the MUR PU. Section IV of RIS 2002-03, Mechanical/Structural/Material Component Integrity and Design," provides information to licensees on the scope and detail of the information which should be submitted to the NRC staff regarding the impact an MUR power uprate has on the structural and pressure boundary integrity of the SSC.

The NRC has recently issued similar MUR PU license amendments for the Surry Power Station, Units 1 and 2, on September 24, 2010, (ADAMS Accession No. ML101750002), for the Prairie Island Nuclear Generating Plant, Units 1 and 2, on August 18, 2010, (ADAMS Accession No. ML102030573), and for the North Anna Power Station, Units 1 and 2, on

October 22, 2009 (ADAMS Accession No. ML092250616).

3.4.3.2 Technical Evaluation

The NRC staff's review focused on the licensee's assessment of the impact of the proposed power on the design-basis analysis of record and covers the structural and pressure boundary integrity of the piping, components and supports which make up the NSSS and the BOP systems. The NRC staff's review also focused on the impact of the proposed power on postulated high-energy break line (HELB) locations and corresponding dynamic effects resulting from the postulated HELB, including pipe whip and jet impingement.

Tables 3-1 and 3-2 of Attachment 1 to the licensee's June 23, 1011, submittal, show the pertinent temperatures, pressures, and flow rates for the eight cases (four cases for Braidwood and Byron Unit 1; four cases for Braidwood and Byron Unit 2) associated with Braidwood and Byron's MUR PU conditions. The licensee evaluated the effects of the proposed MUR PU at a bounding reactor core power level of 3658 MWt which corresponds to the proposed licensed power following the uprate (3645 MWt) plus an uncertainty measurement value of 0.345 percent (approximately 13 MWt). The licensee evaluated four cases of NSSS parameters. Cases 1 and 2 represent an average vessel temperature of 575 ºF, with Case 2 representing an average 5 percent (for Unit 1) and 10 percent (for Unit 2) of SG tubes being plugged. Cases 3 and 4 represent an average vessel temperature of 588 ºF, with Case 4 representing an average 5 percent (for Unit 1) and 10 percent (for Unit 2) of SG tubes being plugged. The evaluations performed by the licensee to demonstrate continued structural and pressure boundary integrity of the aforementioned SSC, at the proposed licensed power conditions, considered the most limiting values of the parameters stipulated in the eight cases, depending on which parameters were used in the AOR for the SSC.

The guidance in Section IV.1.A of RIS 2002-03, identifies certain structure, system and components (SSCs) to be evaluated to determine whether they are able to support the implementation of the proposed licensed power. The evaluations discussed in Section IV of RIS 2002-03 focus on determining what impact the proposed power would have on the AOR for a particular SSC and determine whether the AOR needs to be revised as a result of the proposed licensed power. If the AOR for a particular SSC was performed at conditions which bound which will be present at the proposed licensed power, no further evaluation is required. Furthermore, Section IV.1.B of RIS 2002-03 indicates that for those SSCs whose AOR is affected by implementation of an MUR PU, the licensee should address the following, as they relate to the impact of the proposed licensed power on the AOR: stresses, cumulative usage factors (i.e., fatigue), flow induced vibration (FIV), and changes in temperature, pressure and flow rates resulting from the PU.

The pressure-retaining components which must be evaluated in support of an MUR PU include the following: the RPV, including the RPV nozzles and supports; the reactor vessel intervals (RVIs); the pressure-retaining portions of the control rod drive mechanisms (CRDMs); NSSS piping, pipe supports and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, secondary side internal support structures and nozzles; the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles and the surge line. The licensee has summarized the design codes of record for the above components in Table IV-1.D-1 of its submittal.

Reactor Vessel Structure

The licensee evaluated the effects of the proposed MUR PU on the structural integrity of the RPV in Section IV.1.A.i of Attachment 7 to the submital. The licensee stated that the Byron and Braidwood reactor vessels were previously analyzed with a minimum normal operating inlet temperature of 538.2°F and a maximum normal operating outlet temperature of 620.3°F. The MUR temperature values (538.2°F - 618.4°F) are bounded by the values in the AOR (538.2°F - 620.3°F). The licensee further stated that the NSSS design transients associated with the reactor vessel components remain unchanged for the proposed licensed power.

In the supplement dated February 20, 2012, the licensee indicated that: (1) the uprated RCS design conditions given in Tables 3-1 and 3-2 of Attachment 7 of the submittal provide a T_{avg} range in which the minimum T_{cold} is 541.4°F and the maximum T_{ho}t is 620.9°F; (2) the reactor vessel AOR evaluated a minimum T_{cold} of 538.2°F and a maximum T_{hot} of 620.3 °F; (3) the uprate maximum T_{hot} of 620.9 °F exceeds the maximum T_{hot} evaluated in the reactor vessel AOR; and (4) the MUR PU minimum T_{cold} is bounded by the minimum T_{cold} evaluated in the reactor vessel AOR.

The licensee further stated that normally a reconciliation analysis would be necessary because the uprated maximum T_{hot} is not bounded by the maximum T_{hot} evaluated in the reactor vessel AOR. However, all Braidwood and Byron 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 PU. Therefore, the minimum T_{cold} and maximum T_{hot} evaluated in the reactor vessel AOR bound those of the proposed licensed power when the plant operational limits are taken into consideration.

Based on the licensee's response, the plant operational limits for all Braidwood and Byron units restricting the minimum Tcold to 538.2 °F and the maximum Thot to 618.4 °F, and the fact that the design parameters used in the AOR for the RPV remain bounding, the NRC staff concludes that there is reasonable assurance that the structural integrity of the RPV will be adequately maintained following the implementation of the proposed uprate.

The licensee also discussed the update of the lifting lug interface loads in Section IV.1.A.i of Attachment 7 to the submittal. In response to the staff's question regarding the relevancy of the lifting lugs and the proposed license amendment, in supplement dated February 20, 2012, the licensee stated that: (1) there are three lifting lugs oriented 120 degrees apart around the external side of the reactor vessel closure head which the integrated head package lift rod assemblies attach through a lift rod clevis and clevis pin; and (2) the lifting lug mechanical loads identified for current operating conditions did not change due to the proposed licensed power.

Based on the licensee's response regarding the function of the reactor vessel closure head lifting lugs and the guidance in RIS 2001-03, the NRC staff finds these lifting lugs outside the scope of the NRC's uprate review. Thus, the NRC staff's evaluations and conclusions discussed in this SE are not applicable to the reactor vessel closure head lifting lugs.

<u>Reactor Vessel Internals (RVI) Mechanical Evaluation</u>

The licensee summarized its evaluation of the effects of the proposed licensed power on the structural integrity of the RVIs in Section IV.1.A.ii of Attachment 7 in Reference 1.

The licensee's mechanical evaluation of the RVIs focused on the impact of the proposed licensed power on the design basis loads from a seismic event, LOCAs and FIV. The licensee stated in Reference 1 that the proposed licensed power has no impact on the design basis

seismic, LOCA, and FIV, loads used in the mechanical evaluation of the RVIs. In a letter dated February 20, 2012, the licensee provided further information and confirmed, that the change in T_{cold} and T_{hot} fluid densities, due to the proposed licensed power, is less than 0.1 percent and the current analysis of record remains unchanged. Furthermore, the licensee, in its letter dated February 20, 2012, stated the following:

1. The AOR for the Braidwood and Byron RVIs was performed with conservative gamma heating rates. The proposed licensed power gamma heating rates were verified to remain bounded by the conservative heating rates used in the AOR.

2. 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 PU.

3. All the design loading conditions noted in Section 3.9.5.2 of the Braidwood and Byron 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 PU. The design loads associated with the design of the reactor vessel internals remain bounded by the AOR. The maximum calculated stresses and cumulative fatigue usage factor for the most limiting component of the reactor vessel internals are unaffected by the MUR PU and remain bounded by the AOR.

4. The impact of the proposed MUR PU on the incore instrumentation support structures, including both the upper support columns and the lower support columns, was assessed. The proposed MUR PU conditions are bounded by the design input used in the AOR; thus, the stresses and the cumulative fatigue usage factors in these components remain unchanged from the AOR.

Considering that the design parameters used in the AOR for the RVIs remain bounding and the RVIs continue to meet their design basis acceptance criteria under the conditions of the proposed licensed power, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these components and that there is reasonable assurance that the structural integrity of the RVIs will be adequately maintained following implementation of the proposed licensed power.

<u>Control Rod Drive Mechanisms (CDRMs)</u>

The licensee evaluated the effects of the proposed licensed power on the structural integrity of the CRDMs in Section IV.1.A.iii of Attachment 7 in the submittal.

In the February 20, 2012, supplement, the licensee indicated that the proposed licensed power conditions have no impact on the seismic or LOCA loads used in the AOR of the CRDMs. In addition, the licensee stated that the CRDMs assessment for the proposed licensed power considered all pressure and thermal design transients and load combinations noted in Section 3.9.4 of the Byron and Braidwood UFSAR and concluded that the proposed licensed power has no impact on the AOR of the CRDMs and remain in compliance with the design basis code of record.

Furthermore, the licensee stated, in the submittal, that the proposed licensed power has no effect on the structural qualification of the integrated head package CRDMs seismic support assembly since the revised loads are bounded by the existing design basis loads. Because the licensee shows that the design input parameters used in the AOR for the qualification of the CRDMs are not affected by the proposed licensed power uprate and the CRDMs continue to meet their design basis acceptance criteria under the conditions of the proposed licensed power, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed licensed power on the CRDMs and that there is reasonable assurance that the structural integrity of the CRDMs will be adequately maintained following implementation of the proposed licensed power.

Reactor Coolant System (RCS) Piping and Supports

The licensee evaluated the effects of the proposed licensed power on the structural integrity of the RCS piping and associated supports in Section IV.1.A.iv of Attachment 7 in the submittal.

In the supplement dated February 20, 2012, the licensee indicated that the conditions associated with the proposed licensed power were evaluated to determine the impact on the existing design basis reactor coolant loop (RCL) analysis for the following:

- RCL piping stresses and displacements,
- Primary equipment nozzle loads (RPV inlet and outlet nozzles, SG inlet and outlet nozzles, and RCP suction and discharge nozzles),
- Primary equipment support loads (RPV nozzle supports, SG 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 licensee further stated that:

- The current RCL thermal analysis in the design basis AOR remains bounding;
- The RCL deadweight and seismic analyses remain unaffected by the proposed licesned power because there is no change to the configuration of the RCL piping and supports;
- Primary side NSSS design transients and pressurizer surge line transients are not affected due to the proposed licensed power;
- There is no adverse effect on the fatigue evaluation of the RCL and pressurizer surge line, including the effects of thermal stratification;
- The LOCA hydraulic forcing functions used in the design basis RCL piping AOR analyses remains bounding for the proposed licensed power;
- Thrust and jet impingement forces used in the current AOR remain bounding;
- There are no changes due to the proposed licensed power to the piping or component qualification from the design basis, including: primary equipment nozzles, Class 1 auxiliary piping analysis, and surge line stratification;
- The NSSS component supports, which include the reactor vessel, steam generator, reactor coolant pump, and pressurizer supports, were assessed at the proposed licensed power and were shown to remain bounded by the current design basis AOR; and
- The maximum primary and secondary stresses and the maximum fatigue usage factors associated with the current design basis AOR are applicable to the proposed licensed power.

Because the licensee shows that the RCS piping and associated supports, and NSSS equipment nozzles and supports continue to meet their design basis acceptance criteria under the conditions of the proposed licensed power, the NRC staff concludes that there is reasonable assurance that the structural integrity of the RCS piping and associated supports, and NSSS equipment nozzles and supports will be adequately maintained following implementation of the proposed licensed power.

Balance of Plant (BOP) Piping and Supports

The licensee evaluated the effects of the proposed licensed power on the structural integrity of the BOP piping and associated supports in Section IV.1.A.v of Attachment 7 in the June 23, 2011, submittal. The licensee in its supplement dated February 20, 2012, provided a list of those BOP piping systems that were evaluated for the proposed licensed power conditions, and provided information regarding the methodology used in the evaluation of BOP piping systems.

The licensee stated in its letters dated February 20, and May 16, 2012 (Reference 5), that the AOR design input parameters for the main steam system, SG blowdown system, AFW system, chemical and volume control system, SI system, containment spray system, and circulating water piping remain bounding. The licensee further stated that there was no pressure increase as a result of the proposed licensed power for the BOP piping systems.

The licensee performed a detailed review of those piping systems that the proposed licensed power operating temperature exceeded the CLTP operating temperature by more than one percent. The licensee stated that for piping systems that are currently in compliance with the design code of record, increasing the system temperature by one percent will not affect the acceptability of the piping and its associated support system. The NRC staff considers a one percent screening criterion reasonable to assess the effects of temperature increase in BOP piping systems.

With the exception of heater drain and condensate booster piping systems, all other BOP piping systems passed the 1 percent screening criterion. For the heater drain and condensate booster piping systems, the licensee determined that the proposed licensed power operating temperature was bounded by the temperature condition used in the design basis AOR. In addition, the licensee stated that no pipe or pipe support modifications are required for the MUR PU.

The NRC staff concludes that the licensee's evaluation of the BOP piping and supports under the proposed licensed power conditions acceptable. This acceptance is based on the licensee's demonstration that the design basis requirements associated with the BOP piping systems will continue to be satisfied following the implementation of the proposed licensed power. Because compliance with the criteria stipulated in the design codes of record for the piping system is maintained, the NRC staff concludes that there is reasonable assurance that the structural integrity of the affected BOP piping and supports will be adequately maintained following the implementation of the proposed PU.

Steam Generators (SGs) Structural Evaluation

The licensee evaluated the effects of the proposed licensed power on the structural integrity of the SGs in Section IV.1.A.vi of Attachment 7 in the submittal. The Unit 1 SGs at Braidwood and Byron are BWI, and the Unit 2 SGs are Westinghouse model D-5.

As stated in Section IV.1.A.vi.1.b of Attachment 7 to Reference 1, the licensee performed a structural analysis of Unit 1 SGs to review the impact of the proposed licensed power conditions. In response to the staff's letter dated February 14, 2012, the licensee in the supplements dated February 20, March 30, and May 16, 2012, stated that the primary and secondary side design temperatures and design pressures for the proposed licensed power remain unchanged from the original analysis; however, the proposed licensed power did result in changes to the transient load conditions requiring a reconciliation analysis of Unit 1 SGs. The licensee also provided the maximum stress intensity and the cumulative fatigue usage factors for the critical components of the primary and secondary sides, including nozzles, of the Unit 1 SGs. The licensee verified that these components meet the acceptance criteria of the ASME Code of record for the respective service conditions.

Beca the licensee has demonstrated that the design basis acceptance criteria associated with the most limiting Unit 1 SG components will remain satisfied following the implementation of the proposed licensed power, the NRC staff concludes that there is reasonable assurance that the structural integrity of the Braidwood and Byron,
Units 1, SGs will be adequately maintained following implementation of the proposed licensed power.

As stated in Section IV.1.A.vi.2.b of Attachment 7 to the submittal, the licensee also evaluated the Unit 2 SGs and concluded that the current design basis analysis remains applicable to the proposed licensed power since the design input parameters used in the AOR are equal to or envelop those parameters associated with the uprate.

Because the licensee has demonstrated that the parameters associated with the uprate are equal to or enveloped by the design input parameters used in the current AOR for the Unit 2 SGs, the NRC staff concludes that there is reasonable assurance that the structural integrity of the Braidwood and Byron, Unit 2 SGs will be adequately maintained following implementation of the proposed uprate.

<u>Reactor Coolant Pump (RCP)</u>

The licensee evaluated the effects of the proposed licensed power on the structural integrity of the RCP in Section IV.1.A.vii of Attachment 7 in the submittal.

The licensee evaluated the impact of the proposed licensed power on the structural integrity of the RCPs and concluded that the existing structural analysis of the RCPs remain unaffected because: (1) there were no changes in the RCS design or operating pressure; (2) the MUR proposed licensed power conditions remain bounded by the design parameters used in the original analysis of record; and (3) there are no changes to nozzle or support foot loads for the proposed licensed power.

Because the licensee has demonstrated that the AOR for structural analysis of the RCPs will remain unaffected following the implementation of the proposed licensed power, the NRC staff concludes that there is reasonable assurance that the structural integrity of the RCPs will be adequately maintained following the implementation of the proposed licensed power.

<u>Pressurizer</u>

In its structural evaluation of the pressurizer, summarized in Section IV.1.A.viii of its June 23, 2011 submittal, the licensee stated that: (1) the limiting pressurizer conditions occur when the

RCS pressure is high and the RCS T_{hot} and T_{cold} are low; and (2) no changes were made in the RCS design or operating pressure.

In the supplement dated February 20, 2012, the licensee stated that there is no impact on the pressurizer AOR as a result of the proposed licensed power transient changes; and provided a comparison of pressurizer design parameters for the current operating conditions, the proposed licensed power operating conditions, and the AOR design conditions and concluded that the temperature differential parameters used in the AOR remain bounding for the proposed licensed power conditions.

The licensee also performed an assessment of the structural weld overlay for the pressurizer surge, spray, and safety and relief nozzles. The licensee concluded that the AOR for these components bounds the proposed licensed power conditions and the design basis acceptance criteria remain satisfied at the proposed licensed power.

Because the licensee's evaluations demonstrated that the design parameters used in the AOR for the pressurizer, including all components and nozzles, envelop the proposed licensed power conditions, and the design basis acceptance criteria remain satisfied at the proposed licensed power conditions, the NRC staff concludes that there is reasonable assurance that the structural integrity of the pressurizer will continue to be maintained following implementation of the proposed licensed power.

<u>High-Energy Line Breaks (HELB) and Associated Dynamic Effects</u>

The licensee evaluated the effects of the proposed licensed power on systems classified as high energy to determine whether any changes to the HELB AOR will result from the implementation of the proposed licensed power. This assessment is summarized in Section IV.1.B.vii of Attachment 7 to the June 23, 2011, submittal.

As indicated in the summary of the licensee's assessment, the current AOR for HELB was reviewed to compare the temperatures, pressures and flow rates in high energy piping at the uprated conditions with those in the current AOR. Based on this comparison, the licensee determined that the input parameters used in the current AOR bound those at the uprated conditions. As such, the licensee stated that the proposed licensed power does not result in any new or revised pipe break locations. The licensee also concluded that the dynamic effects evaluations associated with the HELB postulated in the current AOR, including those due to jet impingement and pipe whipping, remain valid at the proposed licensed power conditions.

In the April 27, 2012, supplement, the licensee stated that the uprate evaluations appropriately considered the UFSAR criteria related to high and moderate energy fluid system classification, HELB and moderate energy line crack postulation. The HELB evaluations were performed consistent with the Byron and Braidwood UFSAR by evaluating the high energy systems for potential increases in stress; and no new high or moderate energy systems were added as a result of evaluation at uprated conditions, and no new HELB or moderate energy line crack locations were identified.

During the review, NRC staff became aware of a non-conformance related to the turbine building (TB) HELB analysis. The NRC staff requested the licensee to address the impact of resolution of this nonconformance on the accuracy of the information provided in the submittal. In the August 25, 2011 supplement, the licensee confirmed that: (1) the TB HELB analyses, that supported the proposed licensed power, used thermodynamic assumptions that enveloped

both current licensed thermal power and proposed licensed power conditions; (2) no new HELB locations were identified as a result of proposed licensed power in the piping portions in the auxiliary building including the steam tunnel; (3) the design basis for the turbine building HELB analysis will be maintained; (4) the restoration activities to resolve the nonconformance are being tracked in the Byron and Braidwood corrective action program; and (5) the conclusions stated in the submittal, relative to the HELB analyses remain valid.

The NRC staff concludes that there is reasonable assurance that the proposed licensed power does not result in any new or revised pipe break locations and that the dynamic effects associated with the postulated rupture of piping in the current AOR remain valid. This conclusion is based on the licensee's demonstration that the design input for HELB AOR relative to temperatures, pressures and flow rates will remain bounding under the proposed licensed power; that the piping configuration and seismic stresses are unaffected by the proposed licensed power; and that licensee evaluations of the proposed licensed power demonstrated that piping stresses were not affected as such that resulted in new or revised break locations.

High-Energy Line Break (HELB) Non-conformance

The licensee evaluated the consequences of HELBs inside containment, auxiliary building, and TB with respect to impact on safety-related equipment. Inside the containment, the licensee identified one instance where the qualified lifetime of a specific level transmitter was reduced from 36.2 years to 35.64 years. The licensee's evaluation of high-energy piping in the auxiliary building showed either: (1) no change in the safe SD capability, based upon the existing evaluation using compartmentalization, which limits the effects of a HELB to only one train of safety-related equipment, or (2) in those circumstances where a high-energy line may affect more than one train, the license stated that the operating parameters of high-energy lines in the auxiliary building did not adversely affect the maximum temperature, pressure, or relative humidity.

In the submittal, the licensee stated that it analyzed high-energy pipe breaks for piping with a maximum operating pressure that exceeds 275 psig and the maximum operating temperature that equals or exceeds 200 °F. Additionally, the licensee evaluated for cracks in high-energy piping in which either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200 °F. However, this methodology does not agree with the licensing basis stated in the Byron and Braidwood UFSAR. In the Byron and Braidwood UFSAR, Section 3.6.1.1.1, "Definitions," the licensee defines a high-energy fluid system at these facilities as one where either or both of the following requirements are met:

1. maximum operating temperature exceeds 200 °F
2. maximum operating pressure exceeds 275 psig.

The NRC staff requested that the licensee re-evaluate the effects on plant systems at the proposed power using the current licensing basis definition for HELBs. In a letter dated April 27, 2012, the licensee corrected its definition for high-energy lines used in the MUR to agree with UFSAR, Section 3.6.1.1.1, and confirmed that evaluations for high and moderate energy lines were performed using the classification as stated in the UFSAR. The licensee did not identify any additional high energy break or moderate energy line crack locations at the proposed power condition. The licensee's response sufficiently resolves the staff's concern.

By letter dated August 22, 2011 (ADAMS Accession No. ML112150563), NRC staff stated, in part, that:

> The NRC staff has become aware through the inspection program of a current nonconformance from the current licensing and design basis for the high-energy line break analysis provided in part for review of the MUR power uprate license amendment request. In general, a licensee's corrective action program addresses deviations and nonconformances with most elements of the licensing bases. NRC staff involvement in most of these situations is through the inspection, assessment, and enforcement programs. Provided the licensee is able to correct the problem and restore compliance, nonconformance from the licensing bases are not addressed by a licensing-related process. However, in order to have confidence that the related licensing and design basis information provided in your amendment request will not change and lengthen the review process, the NRC staff requires additional information.

The additional information requested was to discuss the licensee plans for resolving the non-conformance. In particular, the licensee was requested to address the impact of resolution on the accuracy of the information provided to the NRC staff in its June 28, 2011.

By late 2012, the NRC staff became concerned with the progress the licensee was making in resolving the non-conformance. In addition, it was not clear to the NRC staff that the licensee would be able to restore conformance to the licensing and design basis. In a letter dated December 6, 2012 (ADAMS Accession No. ML12271A308), the NRC staff stated, in part,

> The NRC staff has determined that the current analyses of record for the HELB does not bound the requested uprated power level. As a result of this determination, the NRC staff needs additional information to support resolution of the HELB nonconformance and complete a detailed review of the power uprate application. The specific information needed is requested in the enclosure to this letter. However, based on discussions with your staff, it is NRC staff's understanding that the modifications and analysis needed to support the NRC's review is not yet available.
>
> As such, because the needed information is not available to proceed, the NRC staff has suspended the review of the MUR pending the completion of the required modifications and analyses associated with the HELB nonconformance. The NRC staff has also determined that a confirmatory audit may be needed to validate that the resolution scope, including extent of condition, modifications, and analyses bound plant operations at the uprated power level.

By letter dated July 5, 2013 (ADAMS Accession No. ML13186A178), the licensee responded to the requested information and stated that they were ready to support a confirmatory audit on or after July 15, 2013. The NRC conducted the audit July 17 and 18, 2013, and determined, in part, that the resolution of the non-conformance was sufficiently complete to restart the review of the MUR application. The licensee’s July 5, 2013, letter, and the audit confirmed that the licensee was using the Gothic Code instead of the Kitty6 code originally used to analyze the temperature and pressure effects of the TB HELB. By e-mail dated August 28, 2013, the NRC staff requested additional information regarding the TB HELBs and how this was used to develop the input to the Gothic Code. The licensee responded with the additional information on September 5, 2013. The information and the NRC staff’s evaluation is discussed below.

Plant Configuration

In the TB piping layout, the main steam (MS), feedwater and heater drain (HD) systems high energy lines are routed in proximity to the ventilation connections between the turbine and auxiliary building. Therefore a HELB in the MS, feedwater, or HD systems can affect the AB safety-related equipment in the Miscellaneous Electrical Equipment Rooms (MEERs), engineered safety feature (ESF) switchgear rooms, cable spreading (CS) rooms, and emergency diesel generator (EDG) rooms that have ventilation connections to the TB. The ventilation connections would allow the HELB steam to enter the safety-related equipment rooms exposing the equipment to high temperature and humidity as well pressurize the rooms thus creating a differential pressure loading on the walls separating the AB rooms. The ventilation connections are provided with fire dampers, the closing of which is important as it stops the flow of steam into the safety-related equipment rooms. The closure of the dampers also results in a loss of ventilation which causes further room heat-up due to equipment internal heat loads. Some of the non-conformances identified in the previous analysis are as follows:

a) The analysis took credit for the fire dampers to close on high temperature and isolate the AB from the HELB. It erroneously assumed the fusible links melted, closing the damper at the time the local environment reached 165 °F. The Underwriters Laboratories standard for the fusible links identifies a range of allowable times for the damper actuation depending on its surrounding temperature.

b) The analysis did not account for the ventilation airflow paths from the AB rooms of interest. It assumed the rooms were dead-ended with no outflow and therefore limited the amount of HELB M&E released into the room. In reality, there are ducted vent paths exiting the room resulting in greater HELB M&E release into the room.

c) The analysis did not comply with the requirements of single-failure criterion.

The UFSAR, Section 3.6, states:

> "The effects of HELBs in the TB have been evaluated with respect to potential impact on safety-related equipment located in adjoining auxiliary building rooms."

and that:

> "The possible effects associated with the postulated break of piping considered are structural loads due to pressurization, increases in pressure and temperature which could affect environmental qualification of equipment, and damage due to pipe whip and jet impingement"

Modifications

The licensee presented the following plant modifications based on which the revised HELB mitigation strategy is developed:

a) Replaced normally closed backdraft-style dampers by normally open reverse-flow-style HELB dampers which are single failure proof (dual sets of blades) in the ventilation connections to protect the MEERs, ESF switchgear rooms, cable tunnel, DG rooms, CS rooms, and non-ESF switchgear rooms. The new dampers close on high differential pressure in the reverse flow direction, i.e., high differential pressure in the direction of flow of the HELB steam. The dampers are designed to close within 0.5 sec of damper-

specific actuation pressure differences in the reverse direction from normal flow. The new HELB dampers have an integral fire damper. The modifications also included addition of eflectors for protection of the dampers from missiles and jet impingement.

b) Installed new HELB barrier doors at elevation 451-ft, modified existing doors at elevation 401-ft and 426-ft, and reinforced roll-up doors at 401-ft and 426-ft (for Braidwood only) to isolate the safety related equipment rooms from HELB effects.

c) Modified logic by providing auto-transfer of TB control room make-up intake to outside air on detecting a TB high pressure

d) Increased temperature set-point of the fire damper fusible link to allow auto-restoration of ventilation flow and close on fire.

e) Modified ventilation fans operational logic by providing an auto-trip on high differential pressure and auto-restart after a time delay.

f) Modified logic by providing an auto-start of EDG supply fans (on outside air) on EDG start. Modified logic by providing an auto-start of EDG supply fans on high room temperature to ensure EDG availability following a TB HELB even when EDG do not auto-start by a Loss-Of-Offsite Power (LOOP) or Safety Injection (SI) signal

g) Modified logic by providing a high temperature auto-trip of diesel oil storage tank (DOST) rooms exhaust fans.

h) Reinforced divisional block walls separating safety related equipment rooms for applicable load combinations, i.e., HELB pressure load and seismic.

The scope of audit included a review of the revised licensing basis analysis that addresses: (1) the increase in pressure, temperature and humidity due to TB HELB that would affect the environmental qualification of the safety related equipment in the AB rooms, and (2) differential pressure loads applied on the AB walls due to TB HELB. Specifically, the audit included a review of the following: (a) the licensee's methodology for the analysis, (b) the licensee's inputs and assumptions for the analysis, (c) the licensee's analyses based on the most limiting postulated break considering maximum mass and energy release (M&E) releases in combination with the most limiting break locations, (d) the results for the pressure, temperature, and humidity in the rooms of the AB their use for the evaluation of environmental qualification of the safety related equipment installed in the AB rooms, and (e) the results for the differential pressures across the separating walls of the AB rooms and their use in the load combinations for stress analysis of the walls.

The review of postulating break locations and the methodology used for it, and pipe whip and jet impingement analysis due to HELB mentioned in the UFSAR, Section 3.6, were not in scope of this audit.

During the July 17, 2013, audit, the licensee provided a slide presentation of the HELB non-conformance resolution. The licensee subsequently presented the GOTHIC HELB model input data tables for boundary conditions, flow paths, initial conditions, components, forcing functions, control variables, heat sinks. and the output graphs on a personal computer.
During the presentation the licensee verbally provided satisfactory responses to the NRC staff questions with the exception of a global question on reasons for using lumped-parameter

GOTHIC models instead a subdivided models and a question on not postulating certain other break locations and considering the M&E releases from these locations. This issue is discussed below in the section identified as “Mass and Energy Release.”

Methodology Review

The licensee used GOTHIC, Version 7.2a, computer code for the TB HELB analysis. GOTHIC is a state-of-the-art general purpose thermal-hydraulics computer code maintained by Numerical Applications, Inc. (NAI), for the Electric Power Research Institute (EPRI) for performing containment analyses. GOTHIC is qualified under the NAI Quality Assurance (QA) program which conforms to the requirements of 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” with error reporting in accordance with 10 CFR Part 21, “Reporting of Defects and Noncompliance.”

GOTHIC is widely used by the nuclear industry and applications of this code have been previously approved by the NRC staff on a case-by-case basis. The NRC has accepted containment analysis using GOTHIC methodology replacing the previously existing analysis for North Anna 1 and 2, Surry 1 and 2, Kewaunee, and Millstone 2 and 3. The NRC has also accepted containment sub-compartment HELB analysis using GOTHIC methodology for Riverbend Station Unit 1. The NRC has accepted outside containment sub-compartment HELB analysis using GOTHIC methodology for the Advanced Boiling Water Reactor, Units 3 and 4, newly planned reactors by the South Texas Project utility.

GOTHIC HELB Model Review

The licensee developed a lumped-parameter GOTHIC model of the TB and AB together with the ventilation air flow paths connections. The model included the modifications described in above at the MUR PU level. The lumped-parameter GOTHIC modeling approach assumes each walled room in the TB and AB as a separate volume within which the transient values of the thermal-hydraulic parameters do not vary spatially. The licensee analyzed MS lbs of 0.3 ft2, 0.5 ft2 , and 1.4 ft2 areas, 20-inch and 26-inch high density (HD) line breaks, and 30-inch feedwater line breaks. The key assumptions used in the analysis are: (a) M&E release considered maximized transient enthalpies of superheated steam for the MSLBs and constant high enthalpy M&E release from liquid line breaks, (b) maximum differential pressure to close and re-open the new dampers, (c) minimum and maximum closure time considered for the new dampers, (d) high differential pressure trips and time-delay auto-restart modeled for AB fans, (e) high room temperature trips modeled for DOST room fans, and (f) modeled AB room concrete walls, floors and ceilings as heat sinks. The NRC staff finds the licensee’s assumptions to be reasonable and acceptable.

Mass and Energy (M&E) Release

For MSLBs, the licensee used the M&E release data given in Westinghouse report WCAP-10961, “Steamline Break Mass/Energy Releases for Equipment Environment Qualification Outside Containment,” Revision 1, October 1985. The transient M&E release data provides progressively increasing enthalpies with decreasing mass flow rates with respect to time, conservatively maximizing superheat to maximize the effect of M&E releases on environmental qualification of equipment. Additional conservatism is present in the M&E release data because the actual M&E release will be less than the assumed due to piping losses up to the TB. The Westinghouse transient M&E release is based on: (a) maximizing superheat, (b) full-power plus measurement uncertainty, (c) offsite power available, (d) failure of a steam line isolation valve to

close, and (e) split breaks. For Byron and Braidwood, the Westinghouse approach has been accepted by the NRC for maximizing the effect of M&E releases on environmental qualification of equipment located outside containment (see NUREG-0876, Supplement No. 7, "Safety Evaluation Report related to the operation of Byron, Units 1 and 2, Docket Nos. STN 50-454 and STN 50-455," November 1986, page 3-4, and NUREG- 1002, Supplement No. 2, "Safety Evaluation Report related to the operation of Braidwood, Units 1 and 2, Docket Nos. 50-456 and 50-457," October 1986, page 3-18). The licensee analyzed three high pressure MS lb sizes, 1.4 ft2, 0.5 ft2, and 0.3 ft2 areas. The maximum effective MS break flow area considered is 1.4 ft2 because of the integral flow restrictors in the SGs. The licensee justified the M&E release from a 1.4 ft2 area MS line break can conservatively be used for the M&E releases for the 36-inch MSLB. In addition the licensee considered low pressure MS breaks of 42-inch at TB EL. 451-ft (9.1684 ft2 area), 44-inch at TB EL. 451-ft (10.0847 ft2 area), and 24-inch at TB EL. 426 ft (2.7922 ft2 area). The licensee justified that the M&E releases for the liquid line breaks which has more mass flow and no superheat bounds the M&E releases from the low pressure MS breaks.

For HD line (liquid) breaks, the licensee calculated the choked mass release using the equations for the Henry-Fauske critical flow model.

For feedwater line (liquid) breaks the licensee used the pump runout mass flow that flashes steam to atmospheric pressure and credits break isolation times to limit water volumes discharged.

NRC Staff Review and Execution of Selected GOTHIC Models

The NRC staff reviewed the input data in selected electronic GOTHIC files provided by the licensee during the audit. Three MSLB model files, one for each TB elevation, and one FW line break model file were selected for review and execution on the NRC computer using the GOTHIC, Version 7.2a, same as used by the licensee.

On executing these models, the staff obtained the same results for the TB pressures, temperatures, relative humidity, switchgear room temperatures, and relative humidity as shown in licensee documents reviewed during the audit. The maximum differential pressures between the switchgear division 1 and 2 rooms were also consistent with licensee results.

As stated in the UFSAR, the HELB analysis should determine the effects of the M&E release from the postulated break on the structural loads due to pressurization in the AB rooms, and effects of increases in pressure and temperature on the environmental qualification of equipment in the auxillary building. In a request for additional information (RAI) dated August 29, 2013, the licensee was requested to provide the following:

> "Describe in detail the postulated piping failures and their locations utilized for the analyses of M&E release from piping located in the TB that could affect safetyrelated equipment located in adjoining auxiliary building rooms and how this information was used provide input the Gothic analysis. if bounding conditions have been utilized for these analyses identify the piping failures utilized, their bounding M&E and the bounding locations that would envelop the resulting effects on the safety-related equipment located in adjoining auxiliary building rooms. In addition, justify how this/these M&Es and location(s) bound others.

> This justification should include, but is not limited to, consideration of a HE [high energy] release near a HELB damper that would allow pressurization of room while the damper is closing while another room is not yet pressurizing because its damper is farther away from the HE release, thereby creating differential pressure across the wall that separates the two rooms."

In its response dated September 5, 2013, (ADAMS Accession No. ML, the licensee stated that the analysis used closure characteristics of newly installed HELB dampers based on the maximum and the minimum time to close obtained from the test data. The licensee further stated that it used engineering judgment to justify that the differential pressure across the walls in the AB would not exceed the calculated values. The NRC staff concluded that the licensee's reliance on engineering judgement was not acceptable.

During a September 25, 2013, telephone discussion with the licensee to clarify its response, the licensee agreed to supplement its response. In its supplemental response dated October 8, 2013, the licensee referred to the TB HELB jet impingement on auxillary building openings analysis results and has drawn the following conclusion:

> …none of the DG Room, ESF Switchgear Room, or MEER HELB dampers are impacted by, or are within the zone of influence of, a line break jet. Any jet created by a HELB would dissipate prior to reaching a HELB damper. Once the jet dissipates (zone of influence) the high energy fluid in the jet would immediately separate into flashed steam and liquid. The flashed steam would immediately mix with air in the TB area, causing relative humidity and pressure in the TB area to rise. Since air with a relative humidity of less than 100% behaves as an ideal gas, the TB area in front of the separate division dampers (distances in Att. 2[of Reference 3]) would pressurize uniformly. Therefore, the separate divisions' HELB dampers in the TB area would be subjected to the same pressure simultaneously. Additionally, the TB HELB dampers will close before the TB area reaches 100% relative humidity, therefore assuming uniform pressurization is valid for the timeframe being discussed. Due to the uniform pressurization of the TB area in front of the separate division dampers, and that there are no DG Room, ESF Switchgear Room, or MEER HELB dampers affected by a HELB jet, the lumped volume approach used in the GOTHIC analysis is justified.
>
> Since the uniform pressurization of the TB area results in both rooms beginning to pressurize at the same time there is no differential pressure created across the walls separating the rooms due to one room beginning to pressurize before the other.

The licensee's qualitative explanation regarding uniform pressurization of division 1 and 2 switchgear rooms is not quantitatively validated. Based on the licensee's explanation, the NRC determined it could not conclude that high pressure heated steam would simultaneously pressurize both switchgear rooms equally and not produce a differential pressure effect. In addition the NRC determined that, not considering blockages such as presence of large equipment; condenser, turbine, reheater's, etc., introduce further uncertainty in the analysis.

Subsequently, in response to the NRC staff's determination, the licensee in its November 18, 2013, supplement provided quantitative evaluation showing that the newly installed HELB

dampers are not directly impinged by the TB HELBs. In addition, the licensee quantitatively justified that maximum wall differential pressures, used for the structural qualification of the auxillary building divisional walls, derived from the closure characteristics of the newly installed HELB dampers, based on the maximum and the minimum time to close obtained from test data, envelope the differential pressures created due to break locations in the TB with regard to damper location. Regarding the consideration of equipment blockages, the licensee stated there is no large equipment or walls located between adjacent HELB dampers. The licensee therefore determined no impact on the HELB initiated sonic pressure wave propagation through the TB environment due to blockages or obstructions. Based on the licensee's evaluation, the NRC staff concluded that the differential pressure on the auxiliary building divisional walls would not exceed the design capability of the walls.

The NRC staff also determined that the licensee demonstrated significant margins and conservatisms included in the GOTHIC analysis differential pressure calculations due to damper performance and ventilation system operation.

In the December 6, 2012, letter, the NRC staff requested the licensee to summarize the extent of condition review related to the HELB non-conformance. In its July 5, 2013, response, the licensee stated it performed a detailed review of the HELB analyses of other plant structures containing high-energy lines that could impact safety-related equipment. The structures reviewed were:

- Auxiliary building (other than those areas impacted by the TB HELBs),
- MSIV Room/main steam tunnel, and
- Containment building

The extent-of-condition review determined that the HELB analyses supporting these structures have been performed consistent with the current Braidwood and Byron licensing basis. Based on this extent-of-condition review, the licensee concluded that the supporting HELB analyses for the above identified structures were not impacted by the non-conformances identified in the TB HELB analyses.

As stated in previous supplemental letters to NRC staff, dated August 25, 2011 (ADAMS Accession No. ML11255A332) and April 27, 2012 (ADAMS Accession No. ML12121A496), the licensee has indicated that the design basis for the TB HELB (i.e., the qualification of Class 1 E electrical equipment in the identified auxiliary building rooms) are not adversely impacted by a TB HELB and confirmed that the conclusions stated in the original LAR for the MUR LAR as related to the HELB analyses remain valid. The licensee also stated that no new high or moderate energy systems were added as a result of evaluation at MUR PU conditions, and no new HELB or moderate energy line crack locations were identified.

The licensee's evaluation concluded that operation at the proposed power level does not result in any new or revised high or moderate energy-line break locations. As a result, the current high and moderate energy line break analyses bound operation at the higher power level. The licensee concludes that the postulated area high temperatures and pressures resulting from HELBs remain valid at the increase power conditions. Based upon the information and evaluations performed by the licensee to show the effects from a HELB at increase power level are bounded by existing plant analyses, the staff finds the HELB analysis acceptable at the proposed power.

Conclusion

Based on the above review, the NRC staff concludes that: (a) the licensee used an approved methodology for the TB HELB analysis, (b) the GOTHIC inputs and assumptions are conservative, (c) the output results for the pressure, temperature, and humidity in the AB rooms to be used for environmental qualification (EQ) are limiting, and (d) the results of differential pressure analysis across the AB walls are limiting.

Based on the licensee's response in the November 18, 2013 supplement, the NRC staff concludes that the licensee satisfactorily justified that the TB HELB analysis meets the current licensing basis requirements.

Spent Fuel Pool (SFP) Structure (RIS 2002-03, Attachment 1, Section VI.1.D)

The NRC staff requested the licensee to provide information and confirm that, for the expected proposed licensed power conditions, the SFP structure including the SFP liner and the spent fuel racks remain capable of performing their intended design functions and continue to be in compliance with the design basis acceptance criteria. The licensee in its letters dated March 30, May 16, June 26, and September 13, 2012, provided the following information:

1. For the case of a full core offload with loss of one heat exchanger train, the peak SFP bulk water temperature will increase to 166.6°F from the current peak SFP temperature of 162.7°F. The licensee also stated that for the scenario of a full core offload with loss of one heat exchanger train the:

    - Peak temperature was calculated using conservative assumptions (e.g., no evaporation heat loss);
    - Calculated temperature of 166.6 °F is a short term condition;
    - Temperature during a normal refueling with two heat exchangers operable will not peak above 140 °F;
    - SFP temperature alarm is set at 149 °F to alert the operators for an abnormal condition, such as loss of SFP cooling; and
    - Average temperature of the SFP walls and the bottom slab, assuming a 70 °F on the exterior face of the SFP structure, will approximately be 118 °F.

2. A detailed structural evaluation of the SFP structure has been performed, using a bounding temperature of 167 °F, to investigate the MUR power uprate condition. The results of this structural evaluation confirm that the SFP structure is in compliance with the design basis acceptance criteria. Specifically, this evaluation resulted in a maximum rebar stress of 30.3 ksi to be within the allowable limit of 54 ksi and concrete shear stress nearly equal to the ACI [American Cement Institute] code allowable limit (i.e., safety factor of 1.01).

    In its June 26, 2012, submittal (Reference 6), the licensee stated that: (1) at the proposed licensed power, evaluation of the effects of the SFP peak temperature on the SFP rebar stresses used the design methodology and load combinations, consistent with the plant licensing basis requirements, described in the UFSAR, Section 3.8.4, but eliminated conservative evaluation of the thermal moment due to the axial temperature increase; (2) in previous design calculations, including the SFP structural evaluation for the stretch power uprate, the thermal moment induced by the axial temperature increase was conservatively treated as a mechanical moment; and (3) removing this

conservatism resulted in the reduction in the rebar stress from 53.7 ksi, as indicated in the NRC SE for the stretch power uprate, dated May 4, 2001 (Reference 7), to 30.3 ksi.

3. The SFP spent fuel racks were evaluated in the existing analyses of record for a design temperature of 200 °F, which bounds the peak MUR SFP temperature of 166.6 °F.

4. The SFP liner and anchorage system were re-evaluated for a temperature of 167 °F which bounds the proposed licensed power peak SFP temperature of 166.6 °F and the analysis results demonstrated that the stresses were within design basis acceptance criteria. The UFSAR will be updated to reflect the results of the SFP liner re-evaluation for the proposed licensed power condition.

The NRC staff concludes that there is reasonable assurance that the structural integrity of the SFP structure, the SFP liner and racks will be adequately maintained following the implementation of the proposed licensed power. This conclusion is based on the following:

1. The AOR for the SFP racks will remain unaffected following the implementation of the MUR PU as the design of these components were based on a temperature of 200 °F which bounds the peak MUR SFP temperature of 166.6 °F.

2. Section 9.1.2.3.10 of the Braidwood and Byron UFSAR states the following:

   > The liner plate and anchorage system have been designed for the forces resulting from long-term shrinkage of concrete, and a temperature rise to 158ºF from the 70ºF ambient temperature with nominal cooling. The maximum compression force in the liner is calculated using the total strain of the long-term shrinkage of the concrete and the temperature rise. This compressive stress in the liner is limited to 0.90 Fy.

   The licensee in its letter dated September 13, 2012, stated that the AOR for the SFP liner and its anchorage system has been revised to assess the effects of the the proposed licensed power temperature condition. The results of this evaluation, which considered a temperature of 167 °F bounding the peak MUR SFP temperature of 166.6 °F, demonstrated that the liner and its anchorage system are in compliance with the design basis acceptance criteria stated in the Braidwood and Byron Braidwood UFSAR. In addition, the licensee stated that the UFSAR will be updated to reflect the results of this analysis.

3. The licensee performed a structural evaluation of the SFP structure, using a bounding temperature of 167 °F, consistent with the plant licensing basis requirements and demonstrated that the SFP structure will remain in compliance with the design basis acceptance criteria.

4. The ACI 349 limits concrete surface temperature to 150 °F for normal operation or any other long-term periods. For the case of a full-core offload with loss of one heat exchanger train, the peak SFP bulk water temperature will increase to 166.6 °F from the current peak SFP temperature of 162.7 °F. This minimal temperature increase of approximately 4 °F is a short-term condition. The acceptance of the current SFP bulk temperature of 162.7 °F has been documented in the NRC SE for stretch power uprate (Reference 7). The temperature of 166.6 °F, as a short-term temperature condition, will

have negligible effects on the SFP concrete properties considering that the SFP temperature alarm is set to 149 °F which provides an additional measure to alert the operators for an abnormal condition.

Containment Structure Design Basis Pressure and Temperature

The NRC staff requested the licensee to confirm that the design basis pressure and temperatures (normal operating and accident temperature) used in the design of the containment structure and its internal structures remain bounding following implementation of the the proposed licensed power. In response to the staff's letter dated Febraury 14, 2012, the licensee confirmed, in its letter dated February 20, 2012 (Reference 2), that:
(1) the design basis containment pressure and temperature for normal operation are not affected by the the proposed licensed power; (2) for primary system pipe breaks (i.e., LOCAs), the containment peak pressure and temperature for the the proposed licensed power remain bounded by the containment structure design pressure and temperature; and (3) for secondary MSLB, the peak pressure remains bounded by the containment design pressure and there is a very small calculated increase (0.6 °F) in the peak containment air temperature for Unit 1 while Unit 2 remains bounded by the AOR. Furthermore, the licensee stated that the design inputs used in the AOR of the containment internal structures remain bounding following implementation of the the proposed licensed power.

The Section 3.1.2 of this SE concluded that a small increase of 0.6 °F in the Unit 1 containment vapor temperature for an MSLB does not affect the current containment structure peak temperature determined by the LOCA condition.

Based on the licensee's response to the staff's February 14, 2012, letter, as outlined above, and the staff's evaluation of a small increase of 0.6 °F in the Unit 1 containment vapor temperature for an MSLB, the NRC staff concluded that there is reasonable assurance that the structural integrity of the containment and its internal structures will continue to be maintained following implementation of the the proposed licensed power because the proposed licensed power containment pressure and temperature conditions are enveloped by the design basis pressure and temperature used in the design of the containment and its internal structures.

3.4.3.3 Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed licensed power on the structural and pressure boundary integrity of pressure-retaining components and supports and RVIs. Based on its review delineated above, the NRC staff finds the proposed licensed power acceptable with respect to the structural integrity of the SSCs affected by the PU. This acceptance is based on the licensee's demonstration that the regulatory requirements will continue to be satisfied following implementation of the proposed licensed power.

Specifically, the licensee demonstrated that: (1) the structural integrity of most SSCs at Byron and Braidwood stations are not affected by the proposed licensed power, as evidenced by the fact that their AOR are unaffected; (2) those SSCs for which the AOR is affected by the proposed power uprate will continue to satisfy the applicable structural design basis acceptance criteria at the proposed licensed power conditions; and (3) the HELB AOR, including postulated break locations and associated dynamic effects, is unaffected by the proposed licensed power.

Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of these SSCs at Braidwood and Byron will be adequately maintained

and the implementation of the proposed licensed power will not preclude the ability of these SSC to perform their intended functions.