NRR-PMDAPEm Resource

From: Holden, Leslie E.:(GenCo-Nuc) [Leslie.Holden@exeloncorp.com]
Sent: Friday, January 31, 2014 8:42 PM
To: Wiebe, Joel
Cc: Borton, Kevin F:(GenCo-Nuc); Gullott, David M.:(GenCo-Nuc)
Subject: RE: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 5 of 5
Attachments: MUR Package for Proprietary and Factual Error Review - Part 5 of 5 LEH.docx

Joel,

Once again thank you for providing us the opportunity to comment.

Based on our review this afternoon we would like to offer you the following comments for your consideration:

Pages 1 & 2/37 - Grid Stability – The PJM System Stability Study for Braidwood was updated and submitted as an LAR supplement on March 30, 2012. The new analysis did not have any failure scenarios. Please add supplement as in addition to the LAR information and remove references to the three breaker failure scenarios. Suggested revision on the attached.

Page 8/37 - Equipment Qualification Program 3rd ¶ - Since the TB HELB has been satisfactorily addressed we would appreciate if the wording in this paragraph would refer to the SE section where the non-conformance has been addressed and the conclusion that it has been satisfactorily addressed. Appreciate consideration to remove discussion of the Operability Evaluation since they have been closed. (See comments in the attached.)
GEMS is the name of a manufacturer and is not an acronym.

Page 10/37 - §3.4.4.3 – Delete parenthetical statement in the stated commitment. The new PJM analysis does not contain any breaker failure scenarios.

Page 17/37 - §3.4.6.1, SR Valves Technical Evaluation – Revise wording to be consistent with the conclusion of our evaluation that the changes were insignificant (not bounded).

Page 27/37 - Table 3.3.1-6 – the 3rd column heading should be 2 to 8 hours. (Reference LAR supplement dated 8/25/11, Attachment 1, page 5 of 9)

Page 28 & 29/37 - Last ¶ carrying over to top of next page – The number of fire zones was not discussed in the 6/23/11 submittal, but was discussed in the 11/1/11 RAI response (Attachment 1, NRC Request 5, page 16). This is adequately discussed in the following paragraph on page 29.

Page 29 &30/37 - Last ¶ carrying over to top of next page – Discussion of the use of the FP system for non fire related events is not accurate. There are 4 provisions for use of the FP system for non-fire suppression events at each station as address in the reference RAI response (11/1/11, Attachment 1, Response to NRC Question 6, pages 17 & 18). They are as marked in the attached.

Page 32/37 - 1st and 2nd ¶s "Changes to Control Room Controls, Displays, and Alarms" – This commitment is to be implement prior to raising power above CLTP (see deletion on attached). (However, we do not have a problem to implement it as stated.) (For reference see LAR Attachment 5 §VII.2.B pages VII-2 & 3.)

Page 34/37 - "Modifications" - This commitment is to be implement prior to raising power above CLTP (See for reference LAR 6/23/11 Attachment 4 and Attachment 5 §VII.3 page VII-3).

Page 35/37 - §3.6 TS 2.1.1.2 – Revise wording to be consistent with the actual TS change.

Page 36/37 - §4.0 Regulatory Commitments – Revise to be "prior to rising above the current licensed thermal power of 3586.6 MWt" to be consistent with our commitments in LAR 6/23/11 Attachment 4

These comments, as well as other minor editorial comments are indicated in "track changes" in Word "Review" mode on the attached.

Please feel free to give me a call if you have any questions.

Leslie

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From: Wiebe, Joel [mailto:Joel.Wiebe@nrc.gov]
Sent: Friday, January 31, 2014 11:24 AM
To: Holden, Leslie E.:(GenCo-Nuc)
Subject: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 5 of 5

Leslie,

Here is Part 5 of 5 of the MUR package for Proprietary and Factual Error Review.

Joel

Hearing Identifier: NRR_PMDA
Email Number: 1100

Mail Envelope Properties (242AFBA318413548869FC73CD15D0EDA06BD95)

Subject: RE: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 5 of 5
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3.4.4 Electrical Engineering (RIS 2002-03, Attachment 1, Section V.1)

3.4.4.1 Regulatory Evaluation

The regulatory requirements which the staff applied in its review of the application, include GDC 17, "Electric power systems." 10 CFR Part 50, Appendix A, requires, in part, that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety. Conformance to GDC 17 is discussed in the Braidwood and Byron UFSAR, Section 3.1.

Section 50.63 of 10 CFR, "Loss of all alternating current [AC] power," requires, in part, that all nuclear plants have the capability to withstand a loss of all ac power (station blackout (SBO)) for an established period of time, and to recover there from.

Section 50.49 of 10 CFR, "Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants," requires, in part, the licensees to establish programs to qualify electric equipment important to safety, located in harsh environment.

3.4.4.2 Technical Evaluation

The licensee indicated that it developed the request consistent with the guidelines in NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Uprate Applications." The electrical equipment design information is provided in Attachment 7, Section V, of the licensee's submittal.

The NRC staff reviewed the licensee's evaluation of the impact of the proposed licensed power on the following electrical systems/components:

- Grid Stability
- Power Block Equipment (Main Generator, Transformers, Isolated-phase Bus Duct)
- Switchyard
- Alternating Current (ac) Distribution System
- Emergency Diesel Generators (EDGs)
- Direct Current (DC) system
- Station Blackout (SBO)
- Environmental Qualification (EQ) Program

Grid Stability

In Section 3.4.7, Attachment 1, of the submittal, and as supplemented by letter dated March 30, 2012, the licensee provided a summary of the grid stability study to support the proposed licensed power for both Braidwood and Byron.

The grid operator, Pennsylvania, New Jersey and Maryland (PJM) Interconnection, completed a system stability analysis to assess the impact of the proposed licensed power for both Braidwood a~~b~~nd Byron on the stability of generating plants in the Commonwealth Edison (ComEd) control area. This study considered a single-unit trip, loss of the largest unit on the grid, loss of the most critical transmission circuit, and the loss of load. All of the scenarios considered in this study were found to be stable with respect to primary-clearing of the fault and maintenance (prior outage) outage. The breaker failure scenarios were also found stable for Braidwood and Byron ~~except for three breaker failure scenarios for Braidwood~~. ~~The licensee~~

Comment [LEH1]: LAR Supplement March 30, 2012, CL
"The revised Generator Transient Stability Study and Interim Generator Deliverability Study report for the Braidwood Station Units 1 and 2 is provided in Attachment 1 to this letter. The revised study indicated that following the implementation of new ComEd criteria, all of the breaker failure scenarios, considered in this study, were found to be stable."

~~stated that any modification required by PJM Interconnection to address breaker failure scenarios will be completed prior to the MUR PU implementation. The licensee's regulatory commitment to address the proposed modification is discussed below in Section 3.2.4.3 of this SE.~~

ComEd Transmission Planning also completed an assessment of the capability of the grid to ensure adequate post-trip and LOCA voltage levels for both Braidwood and Byron. Power flow simulations were performed using 2012 transmission grid models for a total of four system load conditions.

For Braidwood, the study concluded that the lowest post-contingency switchyard voltage is 349.5 kilovolt (kV) which remains above the minimum required switchyard voltage of 349.2 kV. However, for Byron, in one of the load conditions (a postulated scenario that analyzes a unit trip with the other unit in shutdown condition and with a system load level equal to 75 percent of the 50/50 load forecast), the post-contingency switchyard voltage is lower than the minimum required voltage of 339.8 kV.

The NRC staff questioned the measures the licensee used to resolve the above deficiency regarding Byron's offsite power source voltage requirement assuming a single contingency (N-1 condition), to meet the intent of GDC 17. The licensee, in its letter dated August 25, 2011, stated that this potential low voltage condition is an existing condition and is not related to the MUR PU. The licensee further stated that the contingency scenario is considered acceptable because sufficient transmission system assets such as capacitor banks and other generating assets in the area remain available, which can be used to improve contingency voltage above the minimum required for Byron, when necessary. The staff requested the licensee to explain the actions that are required to be taken to address the single contingency voltage requirement to ensure Byron station has the minimum required post-trip voltage for meeting the design basis requirements. In its response, letter dated April 27, 2012, the licensee stated:

> The Byron Grid voltage analysis (documented in Attachment 10b, "ComEd: 2012 Power Grid Voltage Analysis for Byron Generating Station with MUR Power Uprate," of the LAR) postulated [multiple] scenarios including a conservative scenario that identified a potential issue with low voltage in the Byron switchyard for the 75% of the 50/50 load case (i.e., one Byron unit offline and the trip of the operating unit). This postulated scenario is considered conservative because it does not take credit for other readily available transmission or generating assets.
>
> Subsequent to the completion of the grid voltage analysis discussed above, ComEd performed an additional analysis, "Follow-up Analysis for Byron 2012 Grid Adequacy Study." This action confirmed that there are sufficient transmission assets (i.e., existing capacitors that can be switched on remotely) that will maintain the lowest post contingency voltage above the Byron limit. The Transmission Operator (i.e., PJM) is responsible for implementing non-cost actions (i.e., switching on the existing capacitors or taking other actions) as needed to resolve the voltage concern. In addition, there are other generating assets available in the area and the Byron Sensitivity Study Summary shows that post contingency voltage in the Byron switchyard will remain above the Byron limit if some of these generating assets are placed in service.
>
> PJM and ComEd continuously monitor post-contingency voltages regarding Byron Station. The normal PJM operating protocol is for the Transmission Operator (i.e., PJM) to notify the nuclear power plant operator (i.e., Exelon), through the Transmission Owner

(i.e., ComEd), when the post contingency switchyard voltage is predicted to fall below the Byron Station minimum required voltage. PJM Operating Manuals and Byron Station procedures have been developed to control this operating protocol as follows:

- PJM Manual 39, "Nuclear Plant Interface Coordination," Revision 3, Section 2.4, "Notification and Mitigation Protocols for NPIR [Nuclear Plant Interface Requirements] Voltage Limits," controls the notification and mitigation protocols for nuclear plant voltage limits.

- PJM Manual 3, Transmission Operations," Revision 38, Section 5, "Index and Operating Procedures for PJM RTO [Regional Transmission Operator] Operation," controls the Day-Ahead actions and Real-Time Operator actions for nuclear facilities.

    - If the Day-Ahead studies predict potential low voltage conditions at Byron Station, ComEd would add capacitors as needed to maintain the lowest post contingency voltage above the Byron Station limit and, if necessary, Exelon Generation Company, LLC (EGC) would be informed.
    - If necessary, EGC would authorize PJM to place other generating assets in the area online (i.e., redispatch/off-cost generation) to resolve the potential low voltage conditions.

- Byron Station Abnormal Operating Procedure 0BOA ELEC-1, "Degraded Switchyard Voltage," would be entered if predicted or actual degraded grid voltage conditions exist.

    - 0BOA ELEC-1 directs the Byron Station Operator to perform evaluations based on actual Non-ESF 4 kV bus loading to determine if the predicted post Byron Station unit trip voltage or actual voltage is adequate.

- TSs 3.8.1, "AC Sources Operating," and 3.8.2, "AC Sources-Shutdown," would be entered and the appropriate limiting condition for operation (LCO) applied if the Byron Station Operator determines that the predicted or actual voltage is less than the procedurally specified limit.

The NRC staff reviewed the above response and determined that the licensee has established adequate measures to address the potential low voltage issue (N-1 case) identified in the ComEd Transmission Planning study.

Based on its review, the staff finds that the proposed licensed power will not adversely impact the grid stability for both Braidwood and Byron, and the licensee will continue to meet the requirements of GDC 17.

Power Block Equipment

As a result of the PU, the rated thermal power will increase to 3645 MWt from the previously analyzed thermal power level of 3586.~~5~~6 MWt. The staff has reviewed the capability of the main equipment, such as the main generator, isolated phase bus duct, main step-up

transformers, and auxiliary transformers under the proposed licensed power conditions, as discussed below.

In its letter dated June 23, 2011, Attachment 7, Section V.1.F, the licensee stated that the main generator has a nameplate rating of 1361 Mega~~watt~~Volt-amperes (MVA) (based on 75 psig hydrogen pressure), 0.9 power factor, and 25 kV rated voltage. The licensee provided Generator Capability Curves for both Braidwood and Byron in its letter dated December 9, 2011. In its June 23, 2011, letter the licensee provided calculated Megawatts electric (MWe) for each unit of Braidwood and Byron for the Summer and Winter seasons, with a bounding value of 1294.4 MWe for Byron corresponding to the Winter weather.

The NRC staff reviewed the interim generator deliverability study for Braidwood and Byron Unit 1 and Unit 2 at the proposed licensed power, provided in the licensee's June 23, 2011, letter, Attachment 10a and 10b, respectively. Based on projected maximum real power (MWe) and reactive power (MVARs) dispatch, and review of the generator capability curves, the staff finds that each of the main generator ratings is adequate to support the MUR PU operation.

The licensee stated that each of the Braidwood and Byron Unit 1 and Unit 2 main step-up transformers consist of 2-700 MVA transformers in parallel, totaling 1400 MVA capacity. The 1400 MVA rated capacity is above the main generator 1361 MVA output capability. The NRC staff finds that the main step-up transformers have adequate capacity to carry the main generator full output at the proposed licensed power conditions.

The licensee stated that the isolated phase bus (IPB) duct is rated for 33,000 amperes. The generator rated output at 1361 MVA and nominal voltage of 25 kV is 31,431 amperes. The staff finds that the IPB has adequate rating to operate at the proposed licensed power conditions.

According to the UFSAR, each of the four units has a set of two unit auxiliary transformers (UATs), which supply the unit's normal auxiliary power, are connected to the unit main generator by IPB ducts. The licensee stated that the evaluation of the loading summaries determined that the existing UATs have sufficient capacity with a minimum margin of approximately 32 percent at Braidwood and 25 percent at Byron to support the proposed licensed power without modification. The BOP load increase results in a small increase (<0.5 percent) in the loading of UATs. The staff finds that UATs have adequate capacity for the proposed licensed power conditions.

According to the UFSAR, a set of two system auxiliary transformer (SATs), connected to the 345-kV switchyard, supply required auxiliary power under startup, shutdown, and DBA load conditions. The licensee stated that an evaluation of the connected loading determined that the existing SATs have sufficient capacity with a minimum margin of approximately 32 percent at Braidwood and 25 percent at Byron to support operation at the proposed licensed power conditions and no modification is necessary. The staff finds that the SATs have adequate capacity for the proposed licensed power conditions.

Based on the above, the NRC staff finds that power block equipment have adequate capacity for operation at the proposed licensed power conditions without modification.

Switchyard

According to the UFSAR of Braidwood and Byron, the 345-kV switchyard of each station is configured in a double ring bus configuration, with circuit breakers, disconnect switches, buses,

and other associated equipment. Overhead 345-kV transmission lines distribute power to the various points of the transmission system.

In Attachment 7, Section V.1.G of the submittal, the licensee stated that the transmission lead from the main power transformers to the 345-kV switchyard is capable of carrying the generator full output of each unit and the current to the switchyard is bounded by the generator capability. The licensee's evaluation of the switchyard system demonstrated that a small increase in power output does not significantly impact the switchyard equipment. Based on the licensee's evaluation the NRC staff concluded that the switchyard system analyses bound the uprated conditions.

The NRC staff finds that since the main generator and the main step-up transformer ratings remain unchanged due to the MUR PU, the switchyard equipment ratings are not impacted.

Alternating Current (ac) Distribution System

The ac distribution system is the source of power for the nonsafety-related buses and for the safety-related emergency buses. The system consists of the 6.9 kV, 4.16 kV, 480 volt (V), and 120 V systems. In its June 23, 2011, submittal, Attachment 7, Section V.1.E, the licensee stated that the electrical load changes resulting from the proposed uprate occur primarily at the 6.9 kV buses while the 4.16 kV, 480 V, and 120 V, systems will not see a load increase. The 6.9 kV loads that will be affected by the MUR PU are the condensate pump/condensate booster pump motor, heater drain pump motor, and RCP.

In response to a staff question regarding the extent of changes at the 6.9 kV bus, the licensee, in its letter dated December 9, 2011, stated that the revised brake horsepower values of above motors at MUR PU conditions do not exceed the motor nameplate rating and the existing motors remain unaltered. The licensee concluded that the loading margin after the MUR PU at the 6.9 kV buses is a minimum of 36 percent, which is adequate. Additionally, since the motors have not changed and the overall load change is <0.5 percent of the individual bus load, the load changes do not change the short-circuit ratings of 6.9 kV switchgear, and the motor protective relay settings. The licensee also concluded that the worst-case bus voltages on the 4.16 kV safety-related buses are also not impacted due to MUR PU conditions.

In response to another staff question regarding the effect of LEFM Check-plus system on loading at the low voltage buses and other impact on the power distribution system, the licensee, in its letter dated December 9, 2011, stated that the LEFM CheckPlus system will use 120 V AC nonsafety power sources. Therefore, the licensee concluded that the safety-related power supplies and EDG loading will remain unaffected due to LEFM CheckPlus system addition.

Based on the above, the NRC staff finds that the minor load changes at the 6.9 kV system will not adversely impact the safety-related buses which will continue to support the plant safety related loads for the proposed licensed power conditions.

Emergency Diesel Generators (EDGs)

In Attachment 7, Section V.1.A to the submittal, the licensee stated that the onsite ac power system for each unit consists of two diesel generators (DGs), one for each ESF division. The DGs provide an independent emergency source of power in the event of a complete loss of offsite power. The DG supplies all of the electrical loads required for reactor safe shutdown with

or without a LOCA. The licensee stated that the station electric loads that change as a result of the uprate are not fed from the EDG system and there are no increases to the emergency bus loads supported by the EDGs.

The NRC staff finds that the EDG system is not impacted by the proposed licensed power, and therefore the existing system remains adequate for operation at the proposed licensed power conditions.

Direct Current (dc) System

The dc power system provides control and motive power for vital equipment during all normal as well as emergency conditions of the plant. In Attachment 7, Section V.1.E, to the submittal, the licensee stated that the 125 V dc system loads are not related to the power generation process and are independent of the uprate.

Since, the proposed licensed power does not impact the loading of the dc power system, the staff finds that the 125 V dc electrical distribution system remains acceptable for operation at the proposed licensed power conditions.

Station Blackout (SBO)

Section 50.63 of 10 CFR requires, in part, that each light-water cooled nuclear power plant be able to withstand and recover from a loss of all ac power, referred to as an SBO. Braidwood and Byrons' SBO coping duration is four hours.

In Attachment 7, Section V.1.B, to the submittal, the licensee evaluated the proposed licensed power impact on the alternate ac power source, and the other SBO coping issues: RCS inventory, condensate storage tank inventory, Class 1E~~,~~ battery capacity, ventilation, compressed air, and containment isolation.

Impact on ac Power Source

The ac power source consists of the excess capacity of the running EDG on the non-blacked out unit. The running EDG can be cross-tied to the bus of the same electrical division on the blacked out Unit from the main control room within 10 minutes. The licensee stated that there are no increases to the emergency loads supported by the EDGs as a result of the proposed licensed power. The total loading on the EDG for SBO will remain within the 2000-hour rating of the EDG. Based on the licensee's evaluation the NRC staff concluded that the proposed licensed power will have no negative impact for coping with a SBO event for the required coping period.

Impact on Reactor Coolant Inventory

The licensee stated that the non-blacked-out unit's available EDG provides power to one charging pump per unit, which will provide the water required for maintaining reactor inventory at an adequate level to ensure the core remains covered and natural circulation is not affected.

Impact on Condensate Storage Tank (CST)

The CST provides adequate inventory for decay heat removal (DHR) following a SBO event at the proposed licensed power conditions. In response to a staff question regarding the

inventories during pre- and post- uprate conditions, the licensee in its letter dated December 9, 2011, stated that as per TS Section B 3.7.6, the CST-required capacity is 212,000 gallons for each unit whereas the required condensate inventory per unit for an SBO event is 81,231 gallons under the proposed licensed power conditions. Therefore, the NRC staff finds that the each station's minimum CST inventory is well above the SBO required inventory.

Impact on Class 1E battery capacity

The licensee stated that both Braidwood and Byron, Class 1E, batteries have sufficient capacity to provide adequate power for safe SD loads. The proposed licensed power does not affect any dc powered loads. In response to a staff question, the licensee, in its supplement dated December 9, 2011, confirmed that battery capacity margins remain available, which are in addition to the design factors that are included in the battery sizing calculation (i.e., 1.05 percent for design margin, 1.25 percent for aging, and 1.11 percent for temperature correction). Based on the licensee's evaluation, the NRC staff concludes that there is not a significant impact on the Class 1E battery capacity.

Impact on Ventilation

The licensee performed an evaluation for the heat, ventilation, and air conditioning (HVAC) in the areas containing SBO equipment, such as CR and auxiliary electric equipment rooms, CC water pump and motor-driven AFW pump area, diesel-driven AFW pump area, ESX pump room, RHR pump room and charging pump room, DG rooms, battery rooms, and miscellaneous electric equipment and switchgear rooms. The licensee stated that the temperatures in these areas are unaffected by the uprate, as the heat load in those areas during SBO is not power level dependent. Based on the licensee's evaluation, the NRC concluded that there is not a significant impact on ventilation.

Impact on Compressed Air

The licensee stated that no equipment that needs compressed air for operability has been identified for SBO. Therefore, the licensee concluded that compressed air is not needed. Based on the licensee's evaluation the NRC concluded that compressed air is not impacted.

Containment Isolation

The licensee stated that the uprate does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is not related to the power level. Based on the licensee's evaluation the NRC concluded that containment isolation at current plant conditions remains applicable at the proposed licensed power conditions.

Conclusion

Based on above information, the staff finds that the proposed licensed power will have no adverse impact on Braidwood and Byron's SBO coping duration. Therefore, the NRC staff finds that the licensee will continue to meet the requirements of 10 CFR 50.63 for the proposed licensed power conditions.

Equipment Qualification (EQ) Program

In Attachment 7, Section V.1.C to the submittal, the licensee stated that Class 1E electrical equipment will function, as required, under normal, abnormal, and/or accident environmental conditions. No equipment will be added, removed, or modified as a result of the proposed licensed power. In addition, there is no change in the function of the equipment within the scope of the program. The proposed licensed power does not require any zones to be modified and has no impact on the qualification process.

The evaluation of the effects of the proposed licensed power on the environmental parameters used in qualifying the Class 1E equipment, such as temperature, pressure, and radiation, under normal, abnormal, and/or accident conditions is discussed as follows.
For the normal operating conditions, all the existing values of the environmental parameters remain bounding for proposed licensed power. Because of the conservatism in the current analyses, these analyses remain bounding for the slight increase in normal radiation doses expected for the proposed licensed power conditions.

For the abnormal conditions, the most relevant for EQ is a two-hour loss of ventilation to various auxiliary building areas following a HELB and accompanying loss of offsite power. The staff became aware through the inspection program, of the existence of a non-conformance to the current licensing basis and design basis for the HELB analysis. The staff requested the licensee to provide a discussion on resolving the current non-conformance. In its letter dated August 25, 2011, the licensee stated that 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. The licensee stated that current non-conforming discrepancies were evaluated in recent Braidwood and Byron's Operability Evaluations, and the evaluations concluded that there is assurance that the HELB design basis requirements of protecting Class 1E and safety-related equipment in the identified rooms are maintained. The licensee stated that it will maintain the design basis for TB HELB and return the plant to a condition that does not require the subject Operability Evaluations. The licensee documented in a letter dated November 13, 2013 the physical modifications to resolve the nonconformances and restore the design basis for the TB H ELB have been completed and that the Operability Evaluations pertaining to the TB HELB non-conformance have been closed at both Braidwood and Byron Stations. ~~The restoration activities are being tracked in the Braidwood and Byron Corrective Action Program.~~As discussed in Section X.X.X, "High-Energy Line Break (HELB) Non-Conformance," the NRC staff has concluded that the licensee satisfactorily justified that the TB HELB analysis meets the current licensing basis requirements.

Comment [LEH2]: It may not be necessary to include this level of detail since the Operability Evaluation have been closed.

Comment [LEH3]: Is it possible to close the loop on this based on the HELB non-conformance evaluation?

The licensee stated that the current qualification of Class 1E electrical equipment in the various auxiliary building areas is not adversely impacted by a TB HELB analysis corresponding to proposed licensed power conditions. The NRC staff evaluation of the resolution of the non-conformance is contained in this SE.

The licensee stated that the temperature and pressure values for the containment under accident environmental conditions were revised for the proposed licensed power conditions. The licensee performed an evaluation and determined that all equipment in containment within the scope of the EQ program remains qualified. Only in one case, a slight reduction in qualified life (from 36.2 to 35.64 years) was required for the Byron, Unit No. 2, ~~graphical evaluation modules (~~GEMS~~)~~ containment level transmitters.

Comment [LEH4]: This is a company name not an abbreviation.

The NRC staff asked the licensee to provide a temperature, pressure and radiation profile to demonstrate that adequate margins remain with respect to EQ of electrical equipment in accordance with the Institute of Electrical and Electronics Engineers Standard (IEEE) 323-1974,

“Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” under the worst-case accident conditions at the proposed licensed power conditions. The licensee, in its supplement dated December 9, 2011, provided the following clarifications regarding the margins relating to EQ parameters:

Radiation

The current total integrated dose (TID) includes a 10 percent margin per IEEE-323-1974 requirements. Also, the TID had been analyzed to a power level of ≥102 percent, which bounds the proposed licensed power. Based on the licensee's evaluation the NRC concluded that the TID analysis remains valid.

Pressure

The long-term LOCA M&E release were both re-analyzed for Braidwood and Byron. For the reanalyzed LOCA M&E release cases, the resultant maximum containment pressure of 42.6 psig and 38.26 psig for Units 1 and 2 are less than the peak containment pressures for the current licensing bases of 42.77 psig and 38.36 psig for Units 1 and 2, respectively. The NRC staff finds that the current peak pressure input for EQ is 50 psig, which bounds the reanalyzed peak containment pressure and the current licensing bases pressure for both LOCA M&E release and release cases.

Temperature

The temperature profile considered for EQ of electrical equipment is a composite curve that bounds the results from the MSLB M&E release inside the containment and the LOCA M&E release analyses. Both the long-term LOCA M&E release and the MSLB M&E release were re-analyzed for the proposed licensed power. The EQ evaluation included consideration of margins in peak temperature and post accident operating time (PAOT) in accordance with the IEEE-323-1974, i.e., 15 °F peak temperature and 10 percent in PAOT. The licensee evaluation showed that all EQ electrical equipment is qualified with sufficient temperature margin. The NRC staff finds that the containment temperature profile considered for EQ is bounded by equipment design temperature.

The NRC staff finds that the licensee adequately addressed the effects of the proposed licensed power on the environmental conditions for the qualification of electrical equipment. Therefore, the staff finds that the proposed licensed power will have no significant impact on the licensee's EQ program and that the licensee will continue to meet the requirements of 10 CFR 50.49.

3.4.4.3 Regulatory Commitments

The licensee provided the following regulatory commitment, applicable to the electrical equipment, in Attachment 4 of the submittal:

> [N]on-safety related modifications for the power uprate, including switchyard modifications ~~(to address the breaker failure scenarios discussed in Section 3.1 of this safety evaluation)~~, will be implemented prior to increasing power above 3586.6 MWt.

Comment [LEH5]: As discussed above in our comments for Section 3.1 above the new analysis for Braidwood does not postulate any breaker scenarios.

The NRC staff reviewed the licensee regulatory commitment. The NRC staff notes that this commitment contains an item that is typically controlled via licensee design change control procedures and processes similar to that required by 10 CFR 50, Appendix B, Criterion III. As a result, the NRC staff did not rely on the commitment to determine the acceptability of the proposed licensed power.

3.4.4.4 Conclusion

Based on the technical evaluation provided above, the staff finds that Braidwood and Byron will continue to meet GDC 17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the proposed licensed power acceptable.

3.4.5 Chemical Engineering and Steam Generator Integrity

3.4.5.1 Introduction

The NRC staff has reviewed the submittal, as supplemented by a letter dated December 9, 2011, in accordance with RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," concerning: Coatings in containment (VI.1.B), Flow-Accelerated Corrosion (IV.1.E), Steam Generators (IV.1.A.vi), and the Chemical Volume Control System (IV.1.A.v).

3.4.5.2 Coatings in Containment (RIS 2002-03, Attachment 1, Section VI.1.B)

Regulatory Evaluation

Protective coating systems (Paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside the containment for their suitability for and stability under design basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on: (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs, and (2) RG 1.54, Revision 2, Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants. Specific review criteria are contained in SRP, Section 6.1.2, Protective Coating Systems (Paints) – Organic Materials Review Responsibilities.

Technical Evaluation

The licensee stated that the current licensing basis for containment conditions during a DBLOCA bounds the maximum accident primary containment conditions under MUR PU conditions. Additionally, the licensee stated that coating acceptability was based on the acceptance criteria of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," and NRC RG 1.54, Revision 0. The staff reviewed the licensee's UFSAR and finds that current licensing basis will continue to bound containment conditions under MUR PU conditions. The staff concludes that the licensee has demonstrated with assurance that the coatings will continue to be acceptable following implementation of the proposed licensed power.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed licensed power on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The staff further concludes that the licensee has provided assurance that the current protective coatings will continue to be acceptable following implementation of the

proposed licensed power and will meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the staff finds the proposed licensed power acceptable with respect to protective coatings systems.

3.4.5.3 Flow-Accelerated Corrosion (FAC) (RIS 2002-03, Attachment 1, Section IV.1.E)

Regulatory Evaluation

The FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel are immune to FAC and ~~s~~ significantly reduced in components containing even small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on the system flow velocity, component geometry, fluid temperature, steam quality, oxygen content, and pH. During plant operation, it is not normally possible to maintain all of these parameters in a regime that minimizes FAC; therefore, loss of material by FAC can occur. The licensee stated that the FAC program is based on GL 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning, May 2, 1989 (ADAMS Accession No. ML031200731), the guidelines in the EPRI Report NSAC-202L, Recommendation for an Effective Flow-accelerated Corrosion Program, and the Institute of Nuclear Power Operations (INPO) report INPO EPG-06, Engineering Program Guide - Flow Accelerated Corrosion (FAC). The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC. The staff reviewed the effects of the proposed licensed power on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of damaged components could be made before reaching a critical thickness.

Technical Evaluation

The licensee stated that the FAC program provides a standardized method of identifying, inspecting, and tracking components susceptible to FAC wear in both single- and two-phase flow conditions. Program elements include: FAC susceptibility analysis and modeling; FAC inspection and evaluation; operational experience reviews; and crossover/crossunder main steam piping and moisture separators/reheaters inspections and evaluations. In general, the licensee stated that plant systems are considered susceptible to FAC unless excluded by defined criteria. The criteria includes: material; moisture content; temperature; dissolved oxygen; frequency of system usage; plant-specific operating experience; and industry operating experience.

The licensee utilizes the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code. The CHECWORKS SFA computer code has been used to create unit-specific databases, and once the database has been built, the application is used to perform analysis and data interpretation. The licensee stated that these analytical models result in wear-rate analyses that rank components in order of predicted FAC wear and predicted time to reach minimum allowable wall thickness. The NRC staff reviewed the licensee's submittal and requested additional information concerning the licensee's FAC program to include a sample of components expected to experience the greatest increases in wear at the proposed licensed power conditions, and a comparison of the nominal, current and measured wall thickness, to the thickness predicted by the CHECWORKS SFA FAC model of those sample components.

In its response dated December 9, 2011, the licensee provided a sample list of components that are expected to experience the greatest wear rates prior to and following implementation of the

proposed licensed power. The licensee stated that the components selected were representative of the highest predicted wear rates from the CHECWORKS updates that were preformed for the stretch PU for Byron, Unit No. 1, which historically has operated at the highest power level of the four Braidwood and Byron units. Of the 20 lines provided, 19 lines showed that the predicted wall thickness was more conservative than the measured wall thickness. The licensee stated that the CHECWORKS SFA models will be updated for both Braidwood and Byron's units in order to incorporate the system changes associated with the proposed licensed power. The NRC staff concludes that the licensee design control program is sufficient to control the update to the models.

The NRC staff finds that the current FAC program incorporates adequate conservatism to ensure that components susceptible to FAC are managed appropriately prior to exceeding minimum wall thickness. The staff finds that the updated FAC program, with the incorporated system changes resulting from the proposed licensed power, will provide assurance that components susceptible to FAC will be managed appropriately at the proposed licensed power.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the proposed licensed power on the FAC analysis and concludes that the licensee has adequately addressed the impact of changes in plant operating conditions on the FAC analysis. Additionally, the staff concludes that the licensee has demonstrated the updated analyses will reasonably predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed licensed power. Therefore, the staff finds the proposed licensed power acceptable with respect to FAC.

3.4.5.4 Steam Generator (SG) Tube Integrity (RIS 2002-03, Attachment 1, Section IV.1.A.vi)

Regulatory Evaluation

The SG tubes constitute a large part of the RCPB. As a result, their integrity is important to the safe operation of the facility. The staff's review in this area covered the effects of changes in operating conditions resulting from the proposed licensed power on the SG materials and program. Specific review criteria for meeting the applicable regulations are contained in SRP Section 5.4.2.1, Steam Generator Materials, for the SG materials and SRP 5.4.2.2, Steam Generator Program, for the SG program.

In general, SRP, Section 5.4.2.1, provides recommendations to ensure that: (1) the materials used to fabricate the SG are selected, processed, tested, and inspected to appropriate specifications, (2) the fracture toughness of the ferritic materials is adequate, (3) the design of the steam generator limits the susceptibility of the materials to degradation and corrosion, (4) the materials used in the SG are compatible with the environment to which they will be exposed, (5) the design of the secondary side of the SG permits the chemical or mechanical removal of chemical impurities, and (6) any degradation to which the materials are susceptible (including fracture) is avoided, can be managed through the inservice inspection program, or can be controlled through limits placed on operating parameters. Performing periodic SG inspections will ensure that the integrity of the SG is maintained at a level comparable to that in the original design requirements.

In general, SRP, Section 5.4.2.2, provides recommendations to: (1) ensure that the design of the SG is adequate for implementing a steam generator program, and (2) verify that the steam

generator program will result in maintaining tube integrity during operation and postulated accident conditions. The SG program is intended to ensure that the structural and leakage integrity of the tubes is maintained at a level comparable to that of the original design requirements.

Technical Evaluation

The Units1 and 2 SGs at Braidwood and Byron are different models as each Unit 1 has model BWI Replacement Steam Generators (RSGs) and each Unit 2, has Westinghouse Model D5 SGs. The licensee evaluated the change in operating conditions at the proposed licensed power conditions. In general, the licensee determined that the change in operating conditions at proposed licensed power conditions is small from a steam generator perspective. As a result, the licensee's evaluations showed that either prior analyses were bounding or that the increase in the various calculated values were within acceptance limits.

The NRC staff evaluated the information provided by the licensee. The staff finds that the changes in operating conditions at the proposed licensed power conditions to be small. The new operating temperatures and pressures are typical of those used by other plants with recirculating SGs, which the NRC staff has approved for use. Similar SGs have operated successfully under these conditions. As a result, from a SG materials perspective, the staff concludes that the materials used in the SG remain acceptable, the fracture toughness of the ferritic materials is adequate, the design still limits the susceptibility of the materials to degradation and corrosion, the materials used in the steam generator remain compatible with the environment, the design permits the removal of impurities, and that any degradation that could occur is either avoided or can be managed.

With respect to the SG program, the changes in operating conditions have no effect on the ability to implement the program. As a result, the staff concludes that the design of the SG remains adequate for implementing the SG program. The changes in operating conditions may result in increased susceptibility to degradation and may result in increased degradation growth rates. Although this may occur, the SG program is still adequate (and acceptable) since it requires the licensee to continue to ensure tube integrity for the operating interval between inspections. With respect to the tube repair criteria included in the TSs for the SG program, the NRC staff expects that small changes in operating conditions will have a small effect, if any, on the structural limits for the tubes. Since the tube repair criterion is determined from the structural limit, it may also be slightly affected by the proposed licensed power conditions. The staff concludes that the tube repair criteria remain valid under the proposed licensed power conditions based on staff approval of identical repair criteria at other similarly designed and operated units and the performance-based TS requirement to ensure tube integrity for the operating interval between inspections. As a result of the above, the staff concludes that the SG program remains acceptable for the proposed licensed power conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed licensed power on the Braidwood and Byron, Units 1 and 2 SGs and concludes that the licensee has adequately assessed the continued acceptability of the SG materials and program under the proposed licensed power conditions. The staff has confirmed that the materials in the SG will continue to be adequate to support operation at the proposed licensed power. The staff has also confirmed that the licensee has a program that ensures SG tube integrity, and that the applicability of this program has not changed as a result of the proposed licensed power.

Therefore, the NRC staff finds the proposed licensed power acceptable with respect to SG tube integrity.

3.4.5.5 Steam Generator Blowdown System (SGBS) (RIS 2002-03, Attachment 1, Section VI.1.B)

Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities and, thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The staff's review covered the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criteria for the SGBS are based on GDC 14), "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP, Section 10.4.8, Steam Generator Blowdown System (PWR).

Technical Evaluation

The licensee stated that the blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is related to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the proposed licensed power, the licensee stated that the blowdown required to control secondary chemistry and SG solids will not be impacted by operation at the proposed licensed power. Therefore, the MUR PU will not challenge the design flow rate of 360 gpm for the SGBS system. The licensee stated that SGBS operating temperatures and pressures will decrease and remain bounded by the existing parameters under uprate conditions. Additionally, the licensee stated that operation at the proposed licensed power will not significantly increase the potential for FAC on the SGBS piping and components.

The staff reviewed the licensee's UFSAR and confirmed that the current licensing basis for the SGBS will continue to bound the system post MUR PU implementation.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of operation at the proposed licensed power on the SGBS and concludes that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The staff further concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of 10 CFR Part 50, GDC-14, following implementation of the proposed licensed power. Therefore, the staff finds the proposed licensed power acceptable with respect to the SGBS.

3.4.5.6 Chemical and Volume Control System (CVCS)

Regulatory Evaluation

The CVCS provides a means for: (1) maintenance of programmed water level in the pressurizer (i.e., maintain required water inventory in the RCS, (2) supplying seal-water flow to the RCPs, (3) control of reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration, and makeup, (4) emergency core cooling (part of the system is shared with the ECCS), and (5) provide means for filling, draining, and pressure testing of the RCS.

The NRC staff has reviewed the safety-related functional performance characteristics of the CVCS components. The NRC's acceptance criteria are based on (1) GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, or rapidly propagating fracture, and of gross rupture, and (2) GDC 29, "Protection Against Anticipated Operational Occurrences," requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in SRP Section 9.3.4, Chemical and Volume Control System (PWR).

Technical Evaluation

The licensee stated that the only auxiliary equipment design transients potentially impacted by operation at the proposed licensed power were those transients associated with full-load NSSS design temperatures (Thot and Tcold). The licensee additionally stated that these temperature transients are defined by the differences between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. Since the operating coolant temperatures in the auxiliary systems are not impacted by the proposed licensed power, the licensee stated that the temperature difference between auxiliary systems and the RCS loops is only affected by changes in the RCS operating temperatures.

The design transients assume a full-load NSSS Thot and Tcold of 630 °F and 560 °F, respectively. The licensee stated that these full-load temperatures were selected for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters. The licensee compared the approved range of Thot (608.6 – 620.9 °F) and Tcold (541.4- 555.1 °F) for the proposed licensed power at full-load with the temperatures used to develop the current design transients, and the comparison showed that the proposed licensed power temperatures are lower, which results in less severe design temperature transients. Therefore, the licensee stated that the existing auxiliary equipment design transients are conservative and bounding for the proposed licensed power.

The NRC staff reviewed the UFSAR and confirmed that the design basis for the CVCS remains bounding for MUR PU conditions. The staff also reviewed the licensee's evaluation of Auxiliary Equipment Design Transients concerning CVCS and required additional information concerning the affect of the proposed licensed power on the nitrogen-16 activity in the letdown system. The licensee's response, dated December 9, 2011, stated that the existing design basis reactor water fission product and activation product inventory remained bounding at MUR PU conditions. Since the existing design basis remains bounding, the staff's finds the proposed licensed power acceptable with regard to nitrogen-16 activity.

Conclusion

The NRC staff has reviewed the licensee's evaluations and UFSAR, Section 9.3.4, and has determined that the CVCS will continue to perform its design basis function based on current

design transients bounding the operating conditions post MUR PU implementation. The staff concludes that the licensee has demonstrated that the CVCS will continue to be acceptable and will continue to meet the requirements of GDCs 14 and 29 following implementation of the proposed licensed power. Therefore, the staff finds the proposed licensed power acceptable with respect to the CVCS.

3.4.6 Effect of MUR on Major Components

3.4.6.1 Safety-Related Valves

Regulatory Evaluation

The NRC staff reviewed the licensee's safety-related valves analyses for Braidwood and Byron, Unit Nos. 1 and 2. The staff examined the overall design change and included plant- specific evaluations of Generic Letter(s) (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." The NRC's acceptance criteria for reviewing the safety related valve analysis are based on 10 CFR 50.55a, "Codes and Standards."

Technical Evaluation

In its submittal dated June 23, 2011, the licensee reviewed the impact of the proposed MUR PU on the existing safety-related valves design basis ~~analysis~~. No changes in RCS flow, design, or operating pressure were made as part of the proposed licensed power. The licensee's ~~analysis~~ evaluation concluded that the temperature changes due to the power uprate ~~are bounded by~~have, at most, an insignificant effect on the differential pressures ~~those~~ used in the existing analyses. As a result, none of the safety-related valves required a change to their design or operation as a result of the proposed licensed power. The analyses also confirmed that the existing MSSVs capacity is adequate for overpressure protection at the proposed licensed power conditions and that the existing lift setpoints are unchanged. The NRC staff reviewed the licensee's analysis and determined that none of the safety-related valves required a change to their design or operation as a result of the proposed licensed power.

Comment [LEH6]: LAR Attachment 5, page IV-19
"The evaluations concluded that the temperature changes due to the power uprate have, at most, an insignificant effect on the differential pressures used in the existing analyses."

The licensee also evaluated the impact of the proposed MUR PU on the current air operated valve (AOV) program, GL 89-10 and GL 96-05 motor-operated valve (MOV) program, and GL 95-07 PLTB program. The overall system evaluations concluded that valve function, valve design, operational conditions, thrust, and torque requirements are unaffected by the MUR PU and all valves remain capable of performing their design basis functions. Therefore, no changes are required to the existing AOV, MOV, and PLTB, programs.

Conclusion

Based on the NRC staff's review of the licensee's evaluations, the NRC staff concluded that the performance of existing safety-related valves is acceptable with respect to the proposed licensed power and 10 CFR 50.55a, "Codes and Standards," continues to be met.

3.4.6.2 Safety-Related Pumps

Regulatory Evaluation

The NRC staff reviewed the licensee's safety-related pump analysis. The NRC's acceptance criteria for reviewing the safety-related pumps analysis is based on the requirements in 10 CFR 50.55a.

Technical Evaluation

The NRC staff reviewed the impact of the proposed licensed power conditions on the existing design basis analyses for safety-related pumps. The evaluation showed that there are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect pump performance. The current plant design is considered bounding under the proposed licensed power conditions and requires no modifications to pump systems.

Conclusion

Based on its review of the licensee's analysis, the NRC staff concludes that the performance of existing safety-related pumps is acceptable with respect to the proposed licensed power and 10 CFR 50.55a, "Codes and Standards," continues to be met.

3.4.6.3 Inservice Testing (IST) Program

Regulatory Evaluation

The NRC's acceptance criteria for reviewing the safety-related pumps analysis is based on the requirements in 10 CFR 50.55a. The Code of Record for Braidwood and Byron, Units 1 and 2, is the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2001 Edition through the 2003 Addenda.

Technical Evaluation

In its June 23, 2011, submittal, the licensee described the review of the IST program for safety-related pumps and valves at Braidwood and Byron, Units 1 and 2, for the proposed licensed power. The IST program assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Therefore, the existing IST program will not be impacted by operation at the proposed licensed power.

Conclusion

Based on its review of the ~~the~~ licensee's evaluation, the NRC staff concludes that the IST program will be acceptable, as is, at the proposed licensed power conditions and 10 CFR 50.55a, "Codes and Standards," continues to be met.

3.5 Safety Programs (RIS 2002-03, Attachment 1, Section VII)

Attachment 1, Section VII is intended to ensure that all necessary changes to procedures and modifications to the facilities to support the uprate are completed prior to implementation of the uprate and will adversely affect defense in depth or safety margins.

3.5.1 Radiological Dose Assessment

3.5.1.1 Regulatory Evaluation

This evaluation addresses the impact of the proposed changes on previously analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67 “Accident source term,” as supplemented by accident specific criteria in Section 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." and 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4 of the SRP. The staff used the regulatory guidance provided in the following documents in performing its review:

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”;
- RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents (DBA) at Light-Water Nuclear Power Reactors”;
- SRP 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000;
- NUREG-0800, SRP, Section 6.4, “Control Room Habitability Systems;”
- RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” Revision 1, March 2007;
- RG 1.145: “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”;
- NUREG/CR-2858, “PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”; and
- NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes,” Revision 1, May1997.

The NRC staff also considered relevant information in the Braidwood and Byron’s UFSAR, TSs, and applicable previous licensing actions.

The licensee has accounted for the potential increase in measurement uncertainty should the LEFM system experience operational limitations. The licensee proposes to modify the Technical Requirements Manual (TRM) to add a TRM Limiting Condition for Operation (TLCO). TLCO 3.3.k, "Feedwater Flow Instrumentation," that allows operation at the proposed licensed power for up to 72 hours with an inoperable LEFM system; otherwise, power must be reduced to less than or equal to the current licensed power level of 3586.6 MWt, which corresponds to 98.3% RTP as noted in the TLCO.

In RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” recommends that, to improve efficiency of the NRC staff’s review, licensees requesting an MUR PU should identify existing DBA analyses of record which bound plant operation at the proposed uprated power level. For any DBA for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis. The NRC staff’s review covers the impact of the

proposed licensed power on the results of dose consequence analyses as noted in RIS 2002-03, Attachment 1, Sections II and III.

3.5.1.2 Technical Evaluation

Radiological Consequences of Design Basis Accidents

The NRC approved a full-scope alternative source term (AST) for Braidwood and Byron by letter dated September 8, 2006 (ADAMS Accession No. ML062340420), as Amendment No. 140 to the Braidwood, Units 1 and 2, and Amendment No. 147 to Byron, Unit Nos. 1 and 2. The licensee performed radiological consequence analyses for the following six DBAs to determine CR and offsite exposure.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accidents (FHA)
- Control Rod Ejection Accident (CREA)
- Locked Rotor Accident (LRA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)

The NRC approved a revised AST for Braidwood and Byr~~y~~on by letter dated February 5, 2009 (ADAMS Accession No. ML090230613), as Amendment No. 155 to the Braidwood and Amendment No. 160 to Byron. These amendments approved removing credit for the CR ventilation system recirculation prefilters and reducing the assumed CR unfiltered inleakage in the AST analyses.

LOCA Radiological Consequences Analysis

The NRC staff reviewed the impact of the proposed 1.63 percent PU on the LOCA radiological analysis, as documented in Section 15.6.5 of the Braidwood and Byron UFSAR. In its submittal, the licensee stated that the current LOCA dose AOR for Braidwood and Byron is unaffected by the requested PU because it was performed assuming a power level of 3658 MWt which bounds the proposed licensed power of 3645 MWt. The LOCA analysis assumptions do not include steam mass releases. Therefore, the revised steam mass releases do not affect the LOCA analysis. The NRC staff verified that the existing Braidwood and Byron UFSAR, Chapter 15, radiological analysis source term and release assumptions bound the conditions for the proposed 1.63 percent PU, considering the higher accuracy of the FW measurement instrumentation. The NRC staff finds that, for the proposed licensed power, the radiological consequences of a postulated LOCA continue to meet the dose limits given in 10 CFR 50.67 and GDC 19.

Fuel Handling Accident (FHA) Radiological Consequences Analysis

The NRC staff reviewed the impact of the proposed 1.63 percent PU on the FHA radiological analysis, as documented in Section 15.7.4 of the Braidwood and Byron UFSAR. In its submittal, the licensee stated that the current FHA dose AOR for Braidwood and Byron is unaffected by the proposed licensed power because it was performed assuming a power level of 3658 MWt which bounds the proposed power level of 3645 MWt. The NRC staff verified that the existing Braidwood and Byron UFSAR, Chapter 15, radiological analysis source term and release assumptions bound the conditions for the proposed 1.63 percent power uprate, considering the

higher accuracy of the FW measurement instrumentation. The NRC staff finds that, for the proposed PU, the radiological consequences of a postulated FHA continue to meet the dose limits given in 10 CFR 50.67 and GDC 19, as well as applicable dose acceptance criteria given in SRP, Section 15.0.1.

Control Rod~~e~~ Ejection Accident (CREA) Radiological Consequences Analysis

The NRC staff reviewed the impact of the proposed 1.63 percent power uprate on the CREA radiological analysis, as documented in Section 15.4.8 of the Braidwood and Byron UFSAR. In its submittal, the licensee stated that the current CREA dose AOR for Braidwood and Byron is unaffected by the proposed licensed power because it was performed assuming a power level of 3658 MWt, which bounds the proposed power level of 3645 MWt. The NRC staff verified that the existing Braidwood and Byron UFSAR, Chapter 15, radiological analysis source term and release assumptions bound the conditions for the proposed 1.63 percent power uprate, considering the higher accuracy of the FW measurement instrumentation. The NRC staff finds that, for the proposed PU, the radiological consequences of a postulated CREA continue to meet the dose limits given in 10 CFR 50.67 and GDC 19, as well as applicable dose acceptance criteria given in SRP, Section 15.0.1.

Locked Rotor Accident (LRA) Radiological Consequences Analysis

The NRC staff reviewed the impact of the proposed 1.63 percent PU on the LRA radiological analysis, as documented in UFSAR, Section 15.3.3, of the Braidwood and Byron UFSAR. The current LRA radiological analysis assumes a power level of 3658 MWt, which bounds the proposed power level of 3645 MWt. The licensee stated that the release pathways, and dose conversion factors are unchanged from the current licensing basis. However, in support of its proposed licensed power submittal, the licensee reanalyzed the steam mass release calculations during a LRA. The licensee's calculated steam mass release results are shown in Table 3.3.1-1.

Table 3.5.1-1

Braidwood and Byron Steam Release for LRA

Current		MUR Uprate	
Interval (hours)	**Total Steam Mass Release (lbm)**	**Interval (hours)**	**Total Steam Mass Release (lbm)**
0 to 2	719,000	0 to 2	457,000
2 to 8	1,109,000	2 to 40	3,323,000
8 to 40	2,664,000		
Total	4,492,000	Total	3,780,000

The licensee calculated the vented steam releases from the intact-loop SGs for the 0 to 2 hour time period and the 2 to 40-hour time period for the LRA. After the first two hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 38 hours are required to cool down and depressurize the plant from no-load conditions to the RHR operating conditions. The licensee revised its methodology such that a separate time step from 2 to 8 hour is no longer calculated. The NRC staff finds that this change in methodology has no impact on the LRA radiological dose assessment. The NRC staff also finds that the current LRA radiological consequence analysis assumes steam mass release rates that bound the re-analyzed values. Therefore, the assumptions and parameters for the current LRA radiological consequences analysis are unchanged.

The NRC staff verified that the existing Braidwood and Byron, UFSAR, Chapter 15, radiological analysis source term and release assumptions bound the conditions for the proposed 1.63 percent PU to 3645 MWt, considering the higher accuracy of the FW measurement instrumentation. Therefore, the NRC staff finds that, for the proposed licensed power, the radiological consequences of postulated LRA continue to meet the dose limits given in 10 CFR 50.67 and GDC 19, as well as applicable dose acceptance criteria given in SRP, Section 15.0.1.

Main Steam Line Break (MSLB) Radiological Consequences Analysis

The NRC staff reviewed the impact of the proposed 1.63 percent PU on the MSLB radiological analysis, as documented in UFSAR, Section 15.1.5, of Braidwood and Byron. The current MSLB radiological analysis is based upon the AST methodology. The analysis involves primary coolant radiological source release to the secondary side from the SG and then to the environment. The source terms for equilibrium conditions with 1 % failed fuel are normalized to the TS Dose Equivalent Iodine-131 (DEI-131) limits in the primary coolant, which removes the power dependence from the analysis. The licensee states that the release pathways and dose conversion factors are unchanged from the current licensing basis. However, in support of its proposed licensed power submittal, the licensee revised the Braidwood and Byron steam mass release calculation. Therefore, the licensee recalculated the MSLB accident dose consequence analysis because steam mass release is an assumption in the MSLB accident analysis.

The licensee states that the revised calculation showed that the steam release values used in the current MSLB accident radiological consequence analysis do not bound MUR conditions. Therefore, the licensee revised Braidwood and Byron MSLB accident radiological consequence

analysis using the updated steam release values. The updated steam mass release values are shown in Table 3.1-2.

Table 3.5.1-2

Braidwood and Byron Steam Release for MSLB

Current		MUR Uprate	
Interval (hours)	**Total Steam Mass Release (lbm)**	**Interval (hours)**	**Total Steam Mass Release (lbm)**
0 to 2	442,000	0 to 2	447,000
2 to 8	977,000	2 to 40	3,279,000 ~~3,726,000~~
8 to 40	2,216,000		
Total	3,635,000	Total	3,726,000

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The licensee calculated the vented steam releases from the intact-loop SGs for the 0 to 2-hour time period and the 2 to 40-hour time period for the LRA. After the first two hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 38 hours are required to cool down and depressurize the plant from no-load conditions to the RHR operating conditions. The licensee revised its methodology such that a separate time step from 2 to 8-hour is no longer calculated. The NRC staff finds that this change in methodology has no impact on the MSLB radiological dose assessment. The CR modeling remains the same as described in the Braidwood and Byron current licensing basis. The licensee's limiting calculated MSLB dose results are given in Table 3.5.1-3.

Table 3.5.1-3

Calculated Radiological Consequences for MSLB

Event Condition	Location	Calculated Dose (rem TEDE[1])	Acceptance Criteria (rem TEDE)
Pre-Accident Iodine Spike	EAB	0.145	25[2]
	LPZ	0.083	25[2]
	Control Room	0.580	5.0[2]
Post-Accident Iodine Spike	EAB	0.201	2.5[3]
	LPZ	0.459	2.5[3]
	Control Room	2.845	5.0[2]

[1] Total effective dose equivalent; [2] From 10 CFR 50.67; [3] From SRP 15.0.1

The NRC verified that the revised MSLB accident radiological dose analysis was calculated using the same methodology that was approved for the Braidwood and Byron AST license

amendment. The NRC also verified that, besides the revised steam mass release values, no other assumption and parameters were changed for the MSLB accident dose analysis that were approved for the Braidwood and Byron AST license amendment. Therefore, the NRC staff finds that, for the proposed licensed power of 3645 MWt, the radiological consequences of postulated MSLB continue to meet the dose limits given in 10 CFR 50.67 and GDC 19, as well as applicable dose acceptance criteria given in SRP, Section 15.0.1.

Steam Generator Tube Rupture (SGTR) Radiological Consequences Analyses

In support of its June 23, 2011, submittal, the licensee revised the Braidwood and Byron SGTR accident analysis. The SGTR accident is postulated as a complete severance of a single SG tube. The licensee SGTR radiological consequences analysis evaluated two separate radiological source term cases; an accident initiated iodine spike case and a pre-accident iodine spike case. The accident initiated iodine spike case assumed the primary coolant activity was initially at the long-term TS limit of 1.0 µCi/gm DE I-131 when the event occurs. The accident is assumed to cause the iodine concentration to spike by addition of iodine activity by a factor of 335 times the equilibrium iodine appearance rate for eight hours. The pre-accident iodine spike case assumed the primary coolant iodine concentration was at the current licensing basis value of 60 µCi/gm DE I-131. These assumptions are consistent with Braidwood's and Byron's current licensing basis.

For both cases, all modeling other than the activity source term is the same. The primary-to-secondary coolant leakage of 0.218 gpm per SG goes to the three intact SGs. The activity in the coolant is available for release to the environment through secondary coolant steaming through the SG PORVs. The licensee assumed an iodine partitioning factor of 0.01 in the intact SGs in accordance with RG 1.183 guidance. One hundred percent of the noble gases are assumed to be released. Primary coolant was assumed to pass through the ruptured SG tubes and be available for release to the outside environment by steaming through the ruptured SG PORV. A portion of the rupture flow flashes directly to steam, while the remainder mixes with the secondary coolant for subsequent steaming through the PORV. In these releases, the licensee also postulates the additional presence of iodine activity at the proposed secondary coolant concentration limit of 0.1 µCi/gm DE I-131. The flashing fraction and partitioning fractions for the ruptured SG are in accordance with
RG 1.183 guidance.

In support of its proposed licensed power submittal, the licensee revised the Braidwood and Byron steam mass release calculation. Therefore, the licensee recalculated the SGTR accident dose consequence analysis because steam mass release is an assumption in the SGTR accident analysis. Break flow, flashed break flow, and steam releases from the intact and ruptured SGs were modeled using data from the thermal and hydraulic analyses discussed in Attachment 5a of its June 23, 2011, submittal. The licensee states that the SGTR dose analyses added 10 percent to the mass transfer that was calculated. The 10 percent increase in mass transfers results in approximately a 10 percent increase in the calculated doses. In addition, approximately 5 percent margin was added to the total calculated doses. The RCS and intact SG masses were modeled at the initial values listed in Table 3.5.1-4 of this SE. The CR modeling remains the same as described in the Braidwood and Byron current licensing basis.

Table 3.5.1-4

Braidwood and Byron RCS and SG Mass Data for Steam Release for SGTR

Comment [LEH7]: LAR Attachment 5a, page III-14 Table II-6

	Unit 1	Unit 2
Event	**Liquid Mass (lbm)**	**Liquid Mass (lbm)**
Initial RCS	548,000	488,000
Initial Ruptured SG	103,000~~E5~~	68,900
Minimum Ruptured SG	41,500	>68,900
Initial Intact SGs (total)	310,000	206,000

The licensee's limiting calculated MSLB dose results are given in Table 3.5.1-5.

Table 3.5.1-5

Calculated Radiological Consequences for SGTR

Event Condition	Location	Unit 1 TEDE Dose (rem TEDE[(1)])	Unit 2 TEDE Dose (rem TEDE[(2)])	Acceptance Criteria (rem TEDE)
Pre-Accident Iodine Spike	EAB	3.5	3.7	25[(2)]
	LPZ	0.63	0.69	25[(2)]
	CR	1.7	2.0	5.0[(2)]
Accident Initiated Iodine Spike	EAB	1.8	2.1	2.5[(3)]
	LPZ	0.33	0.41	2.5[(3)]
	CR	0.46	0.56	5.0[(2)]

[(1)]Total effective dose equivalent. [(2)]From 10 CFR 50.67. [(3)]From SRP 15.0.1

The NRC verified that the revised SGTR accident radiological dose analysis was calculated using the same methodology that was approved for the Braidwood and Byron AST license amendment. The NRC also verified that, besides the revised steam mass release values, no other assumption and parameters were changed for the SGTR accident dose analysis that were approved for the Braidwood and By~~y~~ron AST license amendment. Therefore, the NRC staff finds that, for the proposed licensed power of 3645 MWt, the radiological consequences of postulated SGTR continue to meet the dose limits given in 10 CFR 50.67 and GDC19, as well as applicable dose acceptance criteria given in SRP, Section 15.0.1.

Atmospheric Dispersion Estimates

In the February 13, 2006, (ADAMS Accession No. ML060450176), response to an NRC RAI with respect to the February 15, 2005, Braidwood and Byron AST LAR (ADAMS Accession No. ML050560102), the licensee made the commitment to reevaluate the atmospheric dispersion factors (X/Q values) based on the finer wind speed categories provided in the latest appropriate regulatory guidance the next time calculations associated with the dose consequences for the LOCA, MSLB, CREA, LRA, SGTR and FHA were revised. In support of the proposed licensed power submittal, the MSLB and SGTR accident radiological dose analysis were revised. In accordance with the commitment, the licensee performed a reevaluation of the offsite X/Q values for the MSLB and SGTR. The regulatory guidance on finer wind speed categories is

provided in NRC RG 1.23. The use of finer wind speed categories was the only change made in the reevaluation of the offsite X/Q values.

Meteorological Data

As part of its August 25, 2011, supplement, the licensee provided the meteorological data set for years 1994 - 1998 in the form of hourly data formatted for input into the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1), and in the form of a joint wind speed, wind direction, and atmospheric stability frequency distribution for input to the PAVAN atmospheric dispersion computer code (NUREG/CR-2858). The meteorological data originated from the Braidwood and Byron meteorological tower databases and are the same data previously used as a basis for making atmospheric dispersion estimates for use in the DBA dose assessments performed in support of the February 15, 2005, Braidwood and Byron AST LAR. The NRC staff reviewed the meteorological data and found that it provided an acceptable basis for making estimates of atmospheric dispersion in support of the proposed licensed power.

Control Room (CR) Atmospheric Dispersion Factors

The licensee states in its August 25, 2011, supplement, that the CR X/Q values are unchanged and remain consistent with the values noted in the NRC's September 8, 2006, SE for the Byron and Braidwood AST LAR, Table 1, "Byron Units 1 & 2 and Braidwood Units 1 & 2 Control Room Atmospheric Dispersion Factors." The NRC staff determined that the CR X/Q values listed in the September 8, 2006, AST LAR SE are acceptable for use in making DBA dose assessments in support of the proposed licensed power.

Offsite Atmospheric Dispersion Factors

The licensee calculated the new offsite X/Q values for the MSLB and SGTR using the same methods applied in the February 15, 2005, Braidwood and Byron AST LAR. The offsite X/Q values found acceptable for the other analyzed accidents in the September 8, 2006, AST LAR SE remain unchanged. The licensee used guidance provided in RG 1.145 and the PAVAN atmospheric dispersion computer code. Meteorological data from the 5-year period, 1994-1998, was used in the analysis. The format of the PAVAN meteorological input for the X/Q calculations consisted of a joint wind direction, wind speed, and atmospheric stability class frequency distribution. At each station, the outer containment wall and the midpoint between the two reactors were the assumed release points for the exclusion~~ve~~ area boundary (EAB) and low-population zone (LPZ), respectively.

The X/Q calculations utilized 11 wind speed categories as defined in RG 1.23, Revision 1, with the first category identified as "calm." For both stations, the minimum non-calm wind speed was fixed at 0.8 mph. A midpoint was assumed between each of the Regulatory Guide 1.23, Revision 1 wind speed categories (Nos. 2-11) to be inclusive of all monitored wind speeds. Table 2 of the August 25, 2011, supplement, contains the revised wind speed categories used in the analysis.

The licensee notes in the August 25, 2011, supplement, that the offsite X/Q values used in the SGTR analysis are the same as those used in the MSLB analysis. The descriptions of the updated assumptions for the offsite X/Q values for the SGTR are the same as those for the MSLB offsite X/Q values. The bounding set of offsite X/Q values are also the same for both the SGTR and MSLB.

The NRC staff performed a qualitative review of the inputs and assumptions used in the licensee's PAVAN computer calculations and of the resulting X/Q values. The staff calculated comparative X/Q values, and found the results to be similar to the EAB and LPZ X/Q values calculated by the licensee. Therefore, on the basis of this review, the NRC staff determined that the resulting offsite EAB and LPZ X/Q values for the MSLB and SGTR generated by the licensee and presented in Table 3.3.1-6 of this SE are acceptable for use in making DBA dose assessments in support of the proposed licensed power.

Table 3.3.1-6

Byron and Braidwood Bounding Offsite Atmospheric Dispersion Factors (X/Q Values) for the MSLB and SGTR

	X/Q Values (sec/m^3)				
	0 to 2 Hours	**02 to 8 Hours**	**8 to 24 Hours**	**1 to 4 Days**	**4 to 30 Days**
EAB	6.18E-04				
LPZ	1.10E-04	5.13E-05	3.51E-05	1.53E-05	4.68E-06

Comment [LEH8]: LAR Supplement, dated 8/25/2011, Attachment 1, page 5 of 9

3.5.1.3 Conclusion

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs for the proposed 1.63 percent rated thermal power increase at Braidwood and Byron. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the regulatory requirements and guidance identified in Section 3.3.1.1, above. The staff compared the doses estimated by the licensee to the applicable criteria identified. The NRC staff finds that the licensee's estimates of the EAB, LPZ, and CR doses comply with these criteria. The staff further finds there is assurance that Braidwood and Byron as modified by the proposed license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed licensed power is acceptable with respect to the radiological consequences of DBAs.

3.5.2 Fire Protection (FP)

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions, nor will it significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat due to the MUR PU on the plant's safe-shutdown analysis to ensure that SSCs required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire.

3.5.2.1 Regulatory Evaluation

The acceptance criteria for this review are based on: (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC 3 of Appendix A to 10 CFR Part 50, insofar as it requires that [a] SSCs important to safety be designed and located to minimize the probability and effect of fires, [b] non-combustible and heat resistant materials be used, and [c] fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allowed licensees to use a power uncertainty of less than 2 percent in design basis LOCA analyses, based on the use of state-of-the-art feedwater flow measurement devices that provide for a more accurate calculation of reactor power. Appendix K to 10 CFR Part 50 did not originally require that the reactor power measurement uncertainty be determined, but instead required a 2 percent margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error. This type of change is also commonly referred to as an MUR PU.

Attachment 1, Sections II and III of Attachment 1 of RIS 2002-03, recommends that to improve the efficiency of the staff's review, licensees requesting an MUR PU should identify current accident and transient analyses of record which bound plant operation at the proposed uprated power level. For any design basis accident for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis.

3.5.2.2 Technical Evaluation

The licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 3645 MWt against the previously analyzed core power level of 3586.6 MWt. The NRC staff reviewed Attachment 5, MUR Technical Evaluation, Section VII.6.A, "Fire Protection Program," of the submittal. The staff also reviewed the licensee's approved FP program. The review covered the impact of the proposed MUR PU on the results of the safe-shutdown fire analysis and on the effects of the uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips.

The NRC staff's review of the Attachment 5, Technical Evaluation, Section VII.6.A, "Fire Protection Program," to the submittal identified areas in which additional information was necessary to complete the review of the proposed uprate , the licensee responded to the NRC staff RAI as discussed below.

On page VII-5 of Attachment 5 to the submittal the NRC staff noted that Technical Evaluation, Section VII.6.A, "Fire Protection Program," states that,

> …an analysis of the change in combustible loading determined that the overall increase in fire loading is small and does not change the fire load classification of each affected fire zone…

In its submittal dated June 23, 2011, the licensee stated that the uprate conditions result in changes to combustible loading (addition of instrumentation, control and power cable) ~~in 12 fire~~

~~zones in Braidwood Unit 2; seven fire zones in Byron, Unit No. 1, and seven fire zones in Byron, Unit No. 2, that are within the available combustible loading margins~~. The NRC staff reviewed the licensee's assessment of the proposed plant modifications at uprate conditions and the licensee's statement that the modifications will be evaluated in accordance with the ~~the~~ FP license condition and 10 CFR 50.59 processes to determine whether those modifications will require prior NRC approval before implementation.

Comment [LEH9]: This detail was not discussed in the LAR, but was discussed in the 11/1/11 RAI Response, Attachment 1, NRC Request 5, page 16. Discussion in following paragraph cover the number of fire zones.

The NRC staff noted that the licensee was not clear regarding whether there are FP program plant modifications planned (e.g., adding new cable trays, or re-routing of existing cables) as a result of the uprate. By letter dated November 1, 2011, the licensee indicated that the uprate will not involve plant modifications to the fire protection system, or changes to the fire protection program. However, the licensee did state that the LEFM modification will add small amounts of combustible material in the form of instrumentation, control and power cable insulation to 12 Braidwood, Unit 2, fire zones; seven Byron, Unit No. 1, fire zones; and seven Byron, Unit No. 2, fire zones. The Braidwood, Unit 1, LEFM modification was still under development at the time of the submittal. The change in fire loading resulting from the LEFM modification for Braidwood, Unit 1, is intended to be similar to the changes identified for the other three units. In addition, the licensee stated that Section 2.3 of the Fire Protection Report (FPR), currently lists cable insulation as a fire hazard in the affected fire zones and, therefore, the LEFM installation does not introduce a new fire hazard in these fire zones. The licensee stated that the change in combustible loading is well within the available combustible loading margins.

The licensee indicated that the modification to increase the capacity of the SG PORVs for Braidwood and Byron, Unit 1, only should not add any additional combustible material and has no impact on the FP system or program. The other modifications associated with the PU were still under development at the time of the submittal and will be evaluated in accordance with the licensee's design change process required by 10 CFR Part 50, Appendix B, Criterion III.

The NRC staff questioned whether proposed modifications associated with the SGTR and MTO would impact the FP program and the plant's compliance with its FP licensing basis. Further, the staff questioned how the licensee intends to ensure that once developed and implemented, the modifications will not change this impact. In a supplement dated March 30, 2012, the licensee stated that the SGTR and MTO modifications were still being developed. The impact of these modifications on the fire protection program will be evaluated in accordance with Exelon's configuration change process. The licensee stated that per the process, these modifications will be evaluated to assure the changes do not adversely impact the approved FPR and will not adversely impact the ability to achieve and maintain safe shutdown in accordance with the existing license conditions for the units.

In Attachment 5, Section VII.6.A.i, on page VII-6, the licensee states that the fire protection water is utilized to supply the SFP and as cooling to the centrifugal charging pumps. The NRC staff questioned whether there are any other uses of fire water pumps and water for non-fire protection uses. The NRC staff also questioned whether the non-fire suppression use of fire protection water will impact the need to meet the fire protection system design demands.

In its response dated November 1, 2011, the licensee stated that there is no design basis accident or transient (other than fire) that credits the use of the FP system at Braidwood or Byron. However, the licensee stated that there are situations beyond the design basis in which Braidwood or Byron would use the FP system as an alternate water source in EOPs, and other procedures for situations in which the design basis sources of water are unavailable. The licensee identified the following ~~three~~ four provisions for non-fire suppression use of the FP

system: (1) FP water as a backup source of make-up water for the SFP ~~FP water as an alternate water spray to the suppression pool or drywell~~, (2) FP water as an alternate cooling ~~injection~~ source to the ~~reactor pressure vessel~~ centrifugal charging pumps (temporary hoses) if the design basis sources are not available, ~~and~~ (3a Braidwood only) FP water as a backup source ~~of make-up water for the SFP and/or reactor cavity~~ to supply emergency cooling water to station air compressors in the event of loss of non-essential service water, (3b Byron only) FP water as a backup to supply alternate cooling water to the emergency diesel generators (EDG) in the unlikely event that essential service water is not available, and (4) FP water is credited in certain security event scenarios. Further, the licensee stated that the power uprate has no impact on the use of the FP system in these situations.

Comment [LEH10]: See 11/1/11 RAI Response, Attachment 1, NRC Request 6, pages 17 & 18

The licensee's response satisfactorily addresses the NRC staff concerns and based on the licensee's analysis concluded that all the above non-fire suppression uses of fire protection water are beyond design basis and not affected by the proposed uprate.

3.5.2.3 Conclusion

Based upon its review described above, the NRC staff determined that the proposed MUR PU will not have a significant impact on the FP program or post-fire safe-shutdown capability and, therefore, finds the proposed amendment acceptable.

3.5.3 Human Factors

The scope of this review included licensee-identified changes to operator actions, human-system interfaces, procedures, and training needed for the proposed MUR PU and for consistency with the assumptions and results of the revised SGTR and MTO analyses.

3.5.3.1 Regulatory Evaluation

The NRC staff has developed a standard set of questions for human factors reviews of MURs (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4), which were used for this review.

Appendix A to 10 CFR 50, GDC 19, "Control room," states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Equipment at appropriate locations outside the control room shall be provided: (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Section 50.120, "Training and qualification of nuclear power plant personnel" to 10 CFR requires that the licensee establish, implement, and maintain a training program.

Chapter 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training" to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), is used to determine that the training provided, or to be provided, for reactor operators and senior reactor operators will be adequate to provide assurance that all reactor operator qualification requirement items will be met at the time needed.

Chapter 13.5.2.1, “Operating and Emergency Operating Procedures” to NUREG-0800 is intended to provide the criteria used to describe the operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner.

Chapter 13 of the SRP guidance is supplemented by information in the following NUREGs:

- NUREG-0700, “Human-System Interface Design Review Guidelines” Revision 2;
- NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 2; and,
- NUREG-1764, “Guidance for the Review of Changes to Human Actions.”

Chapter 18 of the SRP provides criteria for reviewing the process by which training programs are developed.

Generic Letter (GL) 82-33, “Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability” provides additional clarification regarding Safety Parameter Display Systems, Detailed Control Room Design Reviews, and RG1.97 (Revision 2) - Application to Emergency Response Facilities, Upgrade of emergency Operating Procedures, Emergency Response Facilities, and Meteorological Data.

Regulatory Guide 1.97 describes a method that the NRC staff considers acceptable for use in complying with the agency’s regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants.

Information Notice 97-78, “Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times.”

Regulatory Issue Summary RIS 2007-21, Revision 1, “Adherence to Licensed Power Limits,” addresses a method for ensuring adherence to their maximum thermal power limit

3.5.3.2 Technical Evaluation

The following subsections include the NRC’s technical evaluation of the licensee’s response to the questions in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4, as well as the licensee’s compliance with the other regulatory documents listed in Section 3.3.3.1, above. The NRC staff’s human factors review addresses whether the licensee has adequately considered the effects of the proposed licensed power, and the revised SGTR and MTO analyses on programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC human factors evaluation is conducted to confirm that the licensee has analyzed the effects of the proposed changes and properly concluded that operator performance will not be adversely affected as a result of system and procedure changes made to implement the proposed licensed power.

Operator Actions (RIS 2002-03, Attachment 1, Section VII.1)

In the submittal, the licensee stated that existing operator actions, with the exception of operator response times to mitigate the SGTR, are not affected by the MUR PU and no new manual actions were created. The licensee reviewed the following safety analyses for potential impact: (1) Appendix R, 10 CFR 50; (2) boron dilution; (3) small-break LOCA; (4) Radioactive steam release from a subsystem or component; (5) large-break LOCA; (6) MSLB; (7) main feedwater

break; (8) SGTR; and (9) FHA. The licensee also evaluated operator response times associated with SGTR and MTO scenarios.

Attachment 1, "Evaluation of Proposed Changes," to the June 23, 2011, submittal, describes one modification being installed to support the SGTR and MTO single- failure assumptions which is a manual operator action to locally isolate a manual isolation valve. This manual action is not replacing an automatic function but is simply an equivalent manual action for locally isolating high head safety injection flow into the RCS. The licensee states that this manual action is not time sensitive and that procedure changes will be made to dispatch an operator to the valve location upon identification of a SGTR accident, well in advance of potential need for manual valve isolation. Attachment 5a of the submittal discusses the SGTR and MTO in more detail and includes a table with operator action times demonstrated during SGTR demonstration runs on the Braidwood simulator. Because the manual action is not time sensitive, the NRC staff finds that the licensee's statement is reasonable in Section I.1. A of their submittal that the operator action times assumed in the analysis are conservative relative to actual operator performance.

Based on the licensee's statements in their submittal relating to any impacts of the proposed licensed power on operator actions, the NRC staff concludes that the proposed licensed power will not adversely impact operator actions or their response times. The NRC staff further finds that the statements provided by the licensee do not adversely affect defense in depth or safety margins and meet the relevant regulatory criteria listed in Section 3.5.3.1, above.

Emergency Operating Procedures (EOP) and Abnormal Operating Procedures (AOP)
(RIS 2002-03, Attachment 1, Section VII.2.A.)

In its June 23, 2011, submittal, the licensee stated that the EOP and AOP were reviewed to determine any proposed licensed power impact. The licensee's determined that no changes are required to operator mitigation actions as a result of the MUR PU with the exception of the operator response times noted for mitigation of the SGTR which were discussed above and found to be acceptable because they were not time sensitive. The NRC staff concludes that the neither the proposed licensed power nor the SGTR analysis revision present any adverse impacts to the EOPs and AOPs. This conclusion is based upon the following licensee statement: (1) EOP setpoints and associated operator procedures will be revised to reflect a total core power that bounds the MUR PU, (2) procedure changes and any associated operator training will be completed during the PU implementation and prior to operation above 3586.6 MWt, and (3) SGTR operator action timing assumptions have been validated by demonstration runs on the Braidwood simulator as part of the licensed operator qualification and requalification program at Braidwood and Byron. Implementation of the PU license amendment will include developing the necessary procedures and documents required for maintenance and calibration of the LEFM system. Plant maintenance and calibration procedures will be revised to incorporate maintenance and calibration requirements. The procedure changes will be addressed using standard Exelon procedure updating processes prior to implementation of the proposed licensed power.

The NRC staff finds that the statements provided by the licensee meet the relevant regulatory criteria listed in Section 3.5.3.1, above.

Changes to Control Room Controls, Displays, and Alarms
(RIS 2002-03, Attachment 1, Section VII.2.B)

In its submittal dated June 23, 2011, the licensee described the evaluations performed to identify CR changes in support of the proposed MUR, SGTR, and MTO analysis changes. The licensee identified that changes are required to certain nonsafety-related systems, including minor equipment changes, replacements, and setpoint or alarm point changes. These changes will be made in accordance with the requirements of 10 CFR 50.59, validated and verified in accordance with the Exelon "Configuration Change Control for Permanent Physical Plant Changes," process, and will be implemented ~~prior to implementation of the proposed licensed power and~~ prior to rising above the current licensed thermal power of 3586.6 MWt. This control process change will initiate any necessary changes to the simulator hardware, procedures, and operator training. The proposed licensed power modification will implement the changes that are required to certain non-safety related systems, including CR displays and alarms.

Comment [LEH11]: LAR Attachment 5 §VII.2.B pages VII-2 & 3 as committed to. We do not have a problem to implement it as stated.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the CR and concludes that the proposed changes are not significant and do not present any adverse effects to the operators' functions in the CR. The licensee stated that all modifications to the CR and simulators and providing training on these changes will be completed prior ~~to the proposed licensed power implementation~~to rising above the current licensed thermal power of 3586.6 MWt. The NRC staff finds that the statements provided by the licensee do not adversely affect defense in depth or safety margins and meet the relevant regulatory guidance listed in Section 3.5.3.1, above.

Comment [LEH12]: Same comment as above.

Control Room (CR) Plant Reference Simulator RIS 2002-03, Attachment 1, Section VII.2.C)

In its submittal, the licensee stated that the proposed licensed power is being implemented under the plant modification process administrative controls. As part of this process, simulator modifications will be implemented. The submittal also included statements that these modifications will be evaluated, implemented and tested per Braidwood and Byron plant procedures. Simulator fidelity will be revalidated per Byron and Braidwood-approved procedures. The licensee stated that necessary simulator modifications will be completed in time to support operator training associated with this uprate.

The NRC staff has reviewed the licensee's evaluation of proposed changes to the plant simulator related to the proposed licensed power. The licensee stated that all modifications to the plant simulator and incorporation of these changes into the operator training program will be completed prior to the proposed licensed power implementation. The NRC staff finds that the statements provided by the licensee do not adversely affect defense in depth or safety margins and meet the relevant regulatory guidance listed in Section 3.5.3.1, above.

Operator Training (RIS 2002-03, Attachment 1, Section VII.2.D)

The licensee stated in its June 23, 2011, submittal, that the operator training program requires revision as a result of the proposed licensed power and SGTR MTO analysis. Operator training will be developed and the operations staff will be trained on the plant modifications, TSs, and procedure changes will be implemented per controlled plant procedures prior to operating above the current licensed thermal power of 3586.6 MWt.

The NRC staff has reviewed the licensee's evaluation of the proposed changes to the operator training program. The staff concludes that the proposed changes are appropriate and do not present any adverse effects to the operators' functions in the CR. The licensee stated that training will be provided on these changes prior to the proposed licensed power implementation. The NRC staff finds that the statements provided by the licensee do not adversely affect

defense in depth or safety margins and meet the relevant regulatory guidance listed in Section 3.5.3.1, above.

Modifications (RIS 2002-03, Attachment 1, Section VII.3)

The licensee stated in its submittal that the LEFM system for Braidwood and Byron, Units 1 and 2, will be installed prior to uprate implementation and that other nonsafety-related modifications for the proposed licensed power, including minor equipment changes, replacements, and setpoint or alarm point changes will be implemented prior to uprate implementation. In addition, modifications to support SGTR MTO analysis single failure considerations will be implemented prior to the uprate. Further, the licensee has stated that plant maintenance and calibration procedures will be revised, the plant simulator will be modified for the proposed licensed power conditions, and the changes will be validated in accordance with plant configuration control processes. Maintenance personnel will be qualified on LEFM and operator training will be completed. The licensee stated that all of the above actions will be completed prior to rising above the current licensed thermal power of 3586.6 MWt~~implementation of the proposed licensed power~~. The NRC staff finds that the statements provided by the licensee do not adversely affect defense in depth or safety margins and meet the relevant regulatory guidance listed in Section 3.5.3.1, above.

Comment [LEH13]: LAR 6/23/11 Attachment 4 and Attachment 5 §VII.3 page VII-3.

Temporary Operation above Licensed Full Power Level (RIS 2002-03, Attachment 1, Section VII.4)

Braidwood and Byron General Operating Procedures (BGP 100-3), "Power Ascension," and BwGP 100-3, "Power Ascension 5% to 100%," proactively direct operator actions to maintain licensed power levels at or below 100 percent. The licensee's November 1, 2011, supplement, references the Nuclear Energy Institute (NEI) guidance endorsed in RIS 2007-21, Revision 1, by the NRC for adhering to the licensed thermal power limit. While operating at rated power, the goal of the operator is to achieve a one-hour average less than or equal to maximum allowed (i.e., the maximum thermal power as stated in the plant operating license). The following guidance is in the procedures:

- During full steady state power operation, operators are directed to monitor reactor power using the 10-minute Calorimetric:
    - In the event the 10-minute calorimetric exceeds 100 percent, during steady state operation, operators are directed to initiate actions within 15 minutes to restore the 10-minute calorimetric to less than 100 percent.
    - While temporary power excursions greater than 100 percent power may occur due to reactor coolant system temperature changes, secondary plant efficiency changes, etc., and are allowable, operators are directed that the 1-hour calorimetric should not be allowed to exceed 100 percent.
- For unplanned activities that are expected to cause an increase in reactor power such that the 10-minute cal~~i~~orimetric will be greater than 100 percent, operators are directed to initiate actions such that the 10-minute calorimetric does not exceed 100 percent.

- For unplanned activities that cause the 10-minute calorimetric to exceed 100 percent operators are directed not to wait for the transient to subside, but to take prompt corrective action to limit the time that the 10-minute calorimetric exceeds 100 percent.

The NRC staff finds that the licensee's procedures direct the operator to take action if there is any deviation above the licensed power levels. The NRC staff concludes that this meets the RIS 2002-03 guidance to limit the allowed deviation to a value corresponding to the uncertainty in power level credited by the uprate application.

3.5.3.3 Conclusion

Based on its review as described above, the NRC staff finds that the licensee has adequately considered the impact of the proposed LAR on operator actions, EOPs and AOPs, CR components, plant simulator, and operator training programs. As a result, the NRC concludes that the licensee's activities meet the regulatory requirements and guidance identified in Section 3.3.3.1, above.

3.6 Changes to Technical Specifications, Protection and Emergency Systems Settings (RIS 2002-03, Attachment 1, Section VIII)

Implementation of the measurement uncertainty recapture uprate will result in the following changes to the license and TSs:

- Item 2.C(1) "Maximum Power Level," of the operating licenses will be increased to 3645 megawatts thermal (entire SE);
- TS Section 1.1, will be revised to change the definition of RATED THERMAL POWER (RTP) to increase the value of RTP from 3586.6 to 3645 MWt (entire SE);
- TS Section 2.1.1.1 will be revised to add the ≥ 1.19 limit for the ABB-NV DNB correlation for a thimble cell and a typical cell (Section 3.2.2);
- TS 2.1.1.2~~-~~ will be revised to delete the ≥ 1.30 for the W-3 DNB correlation ~~1.13 limit for the WRB-2 DNB correlation~~ and add the ≥ 1.13 for the ABB-NV correlation and the ≥ 1.18 for the WLOP DNB correlation (Section 3.2.2);
- TS LCO 3.4.1.c will be revised to change the RCS flow rate to ≥ 386,000 gpm (Section 3.2.2);
- SR 3.4.1.3 will be revised to verify that the RCS flow rate is ≥ 386,000 gpm (Section 3.2.2);
- SR 3.4.1.4 will be revised to verify that the RCS flow rate is ≥ 386,000 gpm (Section 3.2.2);
- TS 5.6.5, "Core Operating Limits Report (COLR)" will be revised to add reference WCAP-14565-P-A (Section 3.2.2).

Comment [LEH14]: Revise to match the required TS change

The acceptability of each of these changes is discussed in the assessment above and therefore finds the proposed changes to the license and TSs acceptable.

3.7 Technical Conclusion

The NRC staff confirmed that the licensee provided all information requested by RIS-2002-03. As the methodology used to quantify the uncertainty in the ~~the~~ reactor thermal power uncertainty calculation is consistent with the limitations and conditions in the NRC approved topical reports, the NRC finds that the licensee may apply a reduced margin for ECCS

evaluation consistent with Appendix K of 10 CFR 50. Therefore, the request to uprate the current licensed power from 3586.6 to 3645 MWt and the associated changes to the TSs for Braidwood, Units 1 and 2, and Byron Unit Nos. 1 and 2 are acceptable.

4.0 REGULATORY COMMITMENTS

In Attachment 4 to the submittal, the licensee provided a list of regulatory commitments. The NRC staff concluded that the licensee's statements with respect to completing items prior to rising above the current licensed thermal power of 3586.6 MWt~~to implementation of the power uprate~~ meets the guidance in RIS 2002-03. Therefore, the NRC staff did not rely on these regulatory commitments as the basis for the finding of public health and safety regarding the proposed amendment. The commitments are as follows

Comment [LEH15]: To be consistent with our commitments in LAR Attachment 4

- Limitations regarding operation with an inoperable LEFM system will be included in the TRM.
- The LEFM system for each unit at Byron and Braidwood Stations will be installed prior to uprate implementation
- Plant maintenance and calibration procedures will be revised to address Cameron's maintenance and calibration requirements.
- For Byron Station Units 1 and 2; and Braidwood Station Units 1 and 2; final acceptance of the unit-specific uncertainty analyses will occur after the completion of the commissioning process.
- Nonsafety-related modifications for the PU, including switchyard modifications, will be implemented.
- Modifications to support SGTR MTO analysis single-failure considerations will be implemented.
- Each station's simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes.
- Maintenance personnel will be qualified on LEFM system.
- Operator training will be completed on the LEFM modification and uprated power operations.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (77 FR 28630). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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