April 17, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: DRAFT SAFETY EVALUATION FOR INCREASE IN REACTOR POWER AT BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 (TAC NOS. MA9428, MA9429, MA9426, AND MA9427)

Dear Mr. Kingsley:

By application dated July 5, 2000, as supplemented by letters dated November 27, 2000, December 21, 2000, January 31, 2001, February 20, 2001, March 26, 2001, April 5, 2001, and April 16, 2001, Commonwealth Edison Company requested amendments to the licenses for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The amendments would approve an increase in maximum thermal power from 3411 megawatts-thermal (MWt) to 3586.6 MWt.

Enclosed is a draft copy of the Safety Evaluation that the staff intends to include with the amendments as the technical basis for approving the requests. Because it is a large document, we are sending it prior to the issuance of the amendment in order to permit your staff an opportunity to review it for accuracy, as well as become familiar with the contents.

Please ask your staff to inform me if they find any errors or inconsistencies.

Sincerely,

/RA/

George F. Dick, Jr., Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures: Draft Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-77

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated July 5, 2000, Commonwealth Edison Company (ComEd, the licensee) requested amendments to the licenses for Byron Station, Units 1 and 2 (Byron), and Braidwood Station, Units 1 and 2 (Braidwood) to reflect approval of an increase in maximum thermal power from 3411 megawatts-thermal (MWt) to 3586.6 MWt for each unit. The amendment request proposed changes to both the licenses and technical specifications (TSs). The licensee stated that the power uprate analyses were performed consistent with the guidelines set forth in Westinghouse Energy Systems Report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant." This WCAP methodology, although not formally reviewed and approved by the NRC, was followed by North Anna, Salem, Indian Point 2, Callaway, Vogtle, Turkey Point, and Farley for their core power uprates, and those uprates were found acceptable.

Additional information was provided in the licensee's letters of November 27, 2000, December 21, 2000, January 31, 2001, February 20, 2001, February 28, 2001, March 26, 2001, April 5, 2001 and April 16, 2001. The letters provided clarifying information that did not change the July 5, 2000, application and the initial proposed no significant hazards consideration determination (December 13, 2000, 65 FR 77914).

Subsequent to the date of the amendment requests, ComEd was merged into Exelon Generation Company, LLC (Exelon). By letter dated February 7, 2001, Exelon informed the NRC that it assumed responsibility for all pending NRC actions that were requested by ComEd.

The scope and depth of the staff's review for the Byron and Braidwood power uprate request were based on the safety evaluation supporting the power uprate amendment issued on April 29, 1998, for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The Farley power uprate

safety evaluation incorporated recommendations from the Report of the Maine Yankee Lessons Learned Task Group, and has since been used by the staff as a "template" for subsequent power uprate reviews. The Maine Yankee Lessons Learned Task Group's report is documented in SECY 97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997.

The staff's evaluation of Exelon's request for Byron and Braidwood follows:

2.0 CORE NUCLEAR AND THERMAL-HYDRAULIC DESIGN

The staff evaluated the effect of the proposed power uprate on fuel assemblies. The staff determined that the increase in reactor power will have a negligible impact on fuel rod fretting, oxidation and hydrating of thimbles and grids, fuel rod growth gap, and guide thimble wear. Therefore, the staff concludes that the fuel assemblies would not be adversely impacted by the proposed core power uprate.

The reactor coolant systems (RCSs) at Byron and Braidwood are similar. The licensee's analyses for the power uprate accounted for known differences relating to the installed steam generators (SGs) at Units 1 (BWI replacements) and Units 2 (original D5). Following the core power uprate, the RCS flow per assembly would be slightly higher than in previous analyses. The RCS total flow rate used in the evaluation of all normal and accident conditions would increase slightly to 380,900 gpm from 371,400 gpm. The proposed TS value of 380,900 gpm bounds the value derived by assuming a thermal design flow of 92,000 gpm/loop in each of the four loops plus a 3.5 percent flow measurement uncertainty. This minimum RCS flow, based on maximum analyzed SG tube plugging of up to 5 percent for the BWI SGs and up to 10 percent for the original D5 SGs, would be retained in the TS to assure that a flow rate lower than that reviewed by the NRC will not be used. The acceptability of these changes was evaluated in the staff's review of the plant transient and safety analysis results discussed later.

The licensee used the NRC-approved method to evaluate the departure from nucleate boiling (DNB) design basis for VANTAGE 5/VANTAGE+ fuel. The NRC-approved revised thermal design procedure (RTDP) combines uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation (WRB-2) predictions to obtain the design limit departure from nucleate boiling ratio (DNBR) values. The current RTDP design limit DNBR values are 1.25 for both thimble and typical cells. As a result of the proposed power uprate, the DNBR values will be modified to 1.24 and 1.25, for thimble and typical cells, respectively. The licensee has included additional margin by performing the safety analyses to DNBR limits higher than the design limit. As described below, the safety analysis DNBR limit was revised from 1.40, for both typical and thimble cells, to 1.33, for both typical and thimble cells. The revised limit includes sufficient margin to offset the rod bow penalty and provides additional margin for operating and design flexibility. To support operation at power uprate conditions, the licensee performed DNBR reanalysis to define new core limits, axial offset limits, and anticipated operational occurrence (AOO) acceptability. These are evaluated later. For those analyses of DNBR where the RTDP is not applicable (e.g., hot zero power steamline break, rod withdrawal from low power), the standard thermal design procedure (STDP) was used. For the STDP application, the DNBR limit applied is the correlation limit DNBR with uncertainties mechanistically applied to the calculation input parameters.

The uprated core results in an increase in the core average linear heat rate from 5.45 kW/ft to 5.73 kW/ft, and in the most positive moderator density coefficient from 0.43 Δk/g/cc to 0.54 Δk/g/cc. These increased values, as well as other nuclear parameter changes (e.g., peaking factors, rod cluster control assembly (RCCA) worth, reactivity coefficients, shutdown margin and kinetics), are considered in the revised safety analyses.

2.1 RCS Sampling System

The sampling system allows the licensee operators to take chemistry samples from the gaseous and liquid compartments in the pressurizer, from RCS hot legs in RCS loops 1 and 3, and from RCS cold legs in RCS loops 1 and 4. With the exception of the upper limit of the temperature range for the hot legs which is slightly higher after increasing the core thermal power, all the temperatures for the RCS loops are below their original values. In addition, the design and operating temperatures for the sample heat exchanger are significantly higher and they bound both RCS loop and pressurizer operating temperatures. The licensee concluded, therefore, that the sampling system will not be adversely affected by the power uprate. The staff concurs with the licensee's conclusion.

3.0 ACCIDENT ANALYSIS EVALUATION

In support of the power uprate, the licensee reevaluated the safety analyses for the Byron and Braidwood Stations for operation at a rated thermal power of 3586.6 MWt. The uprate program included the analysis of the large break loss-of-coolant accident (LBLOCA) to specifically address 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and a reanalysis or evaluation of all other aspects of LBLOCA, small-break loss-of-coolant accidents (SBLOCA), non-LOCA accidents, and nuclear steam supply system (NSSS) components. The LBLOCA analysis addressing 10 CFR 50.46 at uprated power conditions was submitted in a separate Byron and Braidwood license amendment request on October 24, 2000. The staff approved the revised LOCA analysis in amendments 118 for Byron, Units 1 and 2, and amendments 112 for Braidwood, Units 1 and 2. The amendments were issued by NRC letter of April xx, 2001.

The majority of the uprate analyses and evaluations were performed in accordance with the current Byron Station and Braidwood Station licensing bases methodologies. However, a number of specific analyses (e.g., the iodine spike factor, LOCA mass and energy release, and feedwater line break calculations) were performed using new or improved methods. The staff will discuss the specific analyses in the appropriate safety analysis section of this safety evaluation (SE). The licensee analyses were performed consistent with the guidelines set forth in Westinghouse report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Plant," dated 1983.

3.1 LOCA Analysis

The licensee identified three loss-coolant accident (LOCA) items which are affected by the power uprate:

- large break LOCA analyses (conformance with 10 CFR 50.46 (b)(1),(2), and (3)),

- small break LOCA analyses (conformance with 10 CFR 50.46 (b)(1),(2), and (3)), and

- long term cooling hot leg switchover/boron precipitation (conformance with 10 CFR 50.46(b)(4) and (5)).

The staff reviewed the licensee's evaluation of these items.

3.1.1 Large Break LOCA (LBLOCA)

By letter dated April xx, 2001, the staff issued amendments 118 for Byron Station, Units 1 and 2, and amendments 112, for Braidwood Station, Units 1 and 2, the NRC approved the performance of the Braidwood and Byron licensing basis LBLOCA analyses using the Westinghouse Best-Estimate LBLOCA analysis methodology described in WCAP-12945-P-A, March 1998. In the supporting SE, the staff accepted the Westinghouse Best-Estimate LBLOCA analysis methodology, as implemented for the Braidwood and Byron plants, based in part upon the licensee's confirmation that the licensee and its vendor(s) have ongoing processes which assure that LBLOCA input values for parameters having an important effect on peak cladding temperature bound the as-operated plant values for those parameters. In its review the staff considered the application of methodology for uprated power levels as well as the previously licensed power levels.

In the April 5, 2001, letter, the licensee confirmed that the analyses also considered both fuel with Zr_4 cladding and with ZIRLO cladding. The analyses calculated PCTs of 2044 °F for Braidwood and Byron Units 1, and 2088 °F for Units 2. The corresponding calculated oxidation values were below 17 percent local oxidation and 1 percent core-wide. These results conform with the criteria given in 10 CFR 50.46 (b)(1),(2), and (3), and are, therefore, acceptable.

3.1.2 Small Break LOCA

The licensee provided small break LOCA (SBLOCA) analysis results for the Braidwood and Byron plants in a letter dated February 20, 2001. The licensee performed the analyses for the uprated power of 3586.6 Mwt using the Westinghouse NOTRUMP SBLOCA analysis methodology described in WCAP-10079-P-A and WCAP-10054-P-A. The analyses also considered both fuel with Zr_4 cladding and with ZIRLO cladding. The licensee showed that the NOTRUMP methodology continues to apply to the Braidwood and Byron plants, by confirming in a letter dated April 5, 2001, that the licensee and its vendor(s) have ongoing processes which assure that SBLOCA input values for parameters having an important effect on peak cladding temperature bound the as-operated plant values for those parameters.

The analyses results presented in the licensee's February 20, 2001, letter, calculated SBLOCA PCTs of 1624 °F for Braidwood and Byron Units 1, and 1627 °F for Units 2. The corresponding calculated oxidation values were below 17 percent local oxidation and 1 percent core-wide. These results conform with the criteria given in 10 CFR 50.46 (b)(1),(2), and (3), and are, therefore, acceptable.

3.1.3 Hot Leg Switchover and Post-LOCA Long term Cooling

During long term cooling with a large cold leg break, emergency core cooling system (ECCS) water is injected into the cold legs. In a typical bounding case, ECCS water injected into the broken cold leg flows out the break. Remaining ECCS water flows into the reactor vessel (RV) downcomer where it maintains downcomer water level at approximately the break elevation with excess ECCS water flowing out the break. The column of water in the downcomer provides the driving force for maintaining a water level in the core and upper plenum.

The NRC and industry have configured acceptable evaluation models to ensure prevention of boric acid precipitation. Consistent with this approach, the licensees previous analyses were based on the following assumptions that the NRC found acceptable:

1. **There is no path for water to leave the upper plenum.**[1] This means core decay heat will heat the core and upper plenum water to the boiling temperature because no water flows through the core to provide cooling. Core decay heat will be removed by vaporizing water with the resulting steam flowing through one or more SGs and out the break.[2] Since the downcomer water level remains constant due to ECCS injection, water removed from the core and upper plenum by steaming will be replaced by water from the downcomer. Further, since the ECCS water, and hence the downcomer water, contain boron, and since no water is removed from the core and upper plenum, these processes will cause boron to accumulate in the core and upper plenum.[3] If this were allowed to continue, sufficient boron may accumulate in the core to cause boron precipitation, with possible plugging of water flow passages leading to core damage. Boron precipitation is prevented by initiating hot leg injection at a rate greater than the steaming rate. This increases the water level in the core and upper plenum and causes water to flow out the bottom of the core and upward through the downcomer, thus flushing boron out of the core and upper plenum.

2. **Water in the core and upper plenum is well mixed by the boiling process.** This assumption means the boron is uniformly distributed in the core and upper plenum water.

3. **The upper plenum collapsed water level is at the level of the bottom of the hot leg flow area at the reactor vessel.** Two opposing effects occur that are not directly addressed. In one, the core and upper plenum fluid density is lower than the

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[1] Two paths may actually exist. In one, steam flowing from the upper plenum into the steam generators (SGs) may contain water droplets. In the other, there may be a flow passage through the gap between the hot leg nozzles and the upper downcomer that could pass both water and steam. The staff has not accepted either as allowable mechanisms for water to be removed from the upper plenum for licensing basis analyses because of insufficient substantiation.

[2] Steam can also flow through the upper head spray nozzles directly into the upper downcomer. There are typically 32 such openings with a total flow area of about 0.2 ft^2. These are located near the reactor vessel flange elevation, significantly higher than the hot leg nozzle gaps, and probably will not pass water.

[3] There is some solubility of boron in steam. This is not considered in licensing basis analyses.

downcomer water density, tending to increase the core and upper plenum water level. In the other, flow friction and boiling dynamics will restrict steam flow from within the core to the break, tending to decrease the core and upper plenum water level.

4. **The bottom of the well mixed core and upper plenum volume is at the level of the bottom of the active fuel.**

5. **There is no heat transfer between the core and the downcomer water.**

6. **The boron concentration limit is the experimentally determined boron saturation concentration with a four weight percent uncertainty factor.**

7. **Decay heat generation rate is 1.2 times the ANS standard for an infinite operating time as required by Section I.A.4. of 10 CFR 50 Appendix K.**

8. **Decay heat generation includes the 1.02 power multiplier identified by Section I.A of Appendix K.**

9. **The containment contains the maximum deliverable water volumes at the maximum allowable boron concentration.**

10. **Potential boron dilution sources, such as from the spray additive tank, are neglected.**

11. **The calculation neglects any elevation of boiling temperature due to concentration of boron in the core.** More boron will remain in solution as the boiling temperature increases.

12. **The barrel/baffle region volume has been neglected.** There are several hundred flow holes of approximately 2 inch diameter which will allow water to move between the core and upper plenum volume and the baffle region volume. Intuitively, this will allow core and baffle region volumes to mix, thus increasing the effective volume where boron is concentrated and increasing the time available before switchover to hot leg injection. The staff has not accepted any proposals to include this volume because an acceptable analysis supporting inclusion of this volume has not been provided.

With the core and upper plenum volume established and the steaming rate and ECCS boron concentration known, calculation of boron concentration rate is straightforward. Time available before hot leg injection must be initiated is calculated from the known boron concentration as a function of time and the allowable boron concentration.

At its presently authorized power level, the Byron and Braidwood licensee calculated the available time as 8.94 hours after occurrence of the worst-case large cold leg loss-of-coolant accident LOCA (see the licensee's submittal of April 5, 2001). The licensee then rounded this value downward to 8.5 hours in the existing licensing basis. The licensee determined that the

requested power increase would reduce this time. Consequently, the licensees made two changes in the approved evaluation model to enable continued use of existing procedures:

1. They added the volume from the top of the lower core plate to the bottom of the active fuel to the previously assumed well mixed core and upper plenum volumes, and

2. They assumed the lower plenum water temperature was 170 °F on the basis of 170 °F ECCS water at the exit of the RHR heat exchangers coupled with an assumed no interaction of steam with ECCS water in the reactor coolant system.

With these two changes and the requested increased power level, the licensee calculated a hot leg switchover time of 8.53 hours, consistent with the existing 8.5 hour requirement. Consequently, the licensees requested that 8.5 hours be retained as the licensing basis requirement to satisfy 10 CFR 50.46. The staff has considered the licensee's request and has reviewed the changes in the licensee's assumptions as discussed in the following paragraphs.

It is well known that relatively high velocities will be developed in the core and upper plenum due to voids generated by boiling during the time of interest here. In high core heat regions, these velocities will have an upward component and liquid elevated by the voids will return via downward flow in regions of lower core heat and along the core periphery. The dynamics of the downward-flowing liquid support a conclusion that there is some liquid flow out of the bottom of the core that will turn and reenter the core. The volume between the top of the lower core plate and the bottom of the active fuel is of relatively small height, a significant flow area exists in this volume for multi-directional flow, increasing boron concentration in the core is a slowly developing process, and there are known conservatisms in calculating boron concentration rate. Consequently, it is not necessary to quantitatively predict these velocities to conclude that the volume between the top of the lower core plate and the bottom of the active fuel can be assumed to participate in the mixing process. Therefore, the staff accepts the licensee's Item 1 change.[4]

In the April 5, 2001, submittal, the licensee stated that water exiting the residual heat removal (RHR) heat exchanger would be at a time-averaged value of approximately 150 °F and assumed a bounding value of 170 °F could be used for purposes of the boron concentration analysis.[5] They further assumed this temperature water would propagate unchanged into the downcomer and lower plenum.

Assuming the lower plenum water temperature is 170 °F also assumes adiabatic conditions between the RHR heat exchanger and the lower plenum. There are two potential challenges to that assumption:

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4 The licensee stated that experiments have established that the lower head volume is also fully mixed. It did not provide information to substantiate this statement and it did not assume this behavior in its analysis.

5 The time-averaged value is equal to the area of the temperature versus time curve from initiation of recirculation to initiation of hot leg injection divided by the time elapsed between those two conditions.

(a) Heat can flow from the core and upper plenum regions into the downcomer water through the core-former structure or through the wall separating the upper downcomer from the upper plenum, and

(b) Steam flowing in the path from the upper plenum to the break can heat incoming water.

Item (a) is not a substantial concern. Any heat transfer from the core and upper plenum region transfers heat to the incoming water while cooling the core and upper plenum. It makes little difference whether the cooling is provided via this path or via cool water entering the core from the lower plenum. The effect on reducing the boron concentration rate is approximately the same.

Item (b) requires further discussion. Water injected into the broken cold leg is assumed to be lost out the break and whether or not it reacts with steam is irrelevant since there will be no effect on the temperature of water reaching the downcomer. Similarly, water injected into the remaining cold legs that is in excess of water boiloff rate will flow out the break and it is again irrelevant whether or not it interacts with steam in the vicinity of the break. However, steam that interacts with water in the unbroken cold legs or in the upper downcomer could challenge the licensee's 170 °F assumption. The staff addressed each of the significant considerations in the following paragraphs.

The licensee calculated that minimum ECCS and minimum cooling capability would provide a time-averaged RHR heat exchanger outlet temperature of approximately 150 °F. Thus, there is a 20 °F margin to account for steam heating of the incoming water. If all of the steam leaving the upper plenum flows to the cold leg break via the SG in the broken reactor coolant system (RCS) loop, none of it flows to cold legs via unbroken loop SGs, and none of it enters the upper downcomer via the broken loop cold leg, then the only way for steam to interact with incoming water is steam from the reactor vessel upper head spray nozzles. The licensee calculated that the average steam flow rate out of the core is approximately 40 lbs/sec, heatup of the incoming ECCS water from 150 °F to 170 °F could condense about 33 percent of this steam, and the steam flow rate through the spray nozzles would be 12% of the total steam generation rate. It is not clear if the licensee included superheat from the SGs in the calculation.[6] If it did not, then the licensee's steam energy estimate may be low. However, this will not change the conclusion: The licensee's 170 °F assumption remains valid if the only steam interaction is due to steam flow through the spray nozzles.

If steam were to flow from the vicinity of the break into the downcomer, the licensee argues that an air layer would build up that would tend to separate the steam from the ECCS water, thus reducing the steam condensation rate. If this "incoming" source of steam is greater than "outgoing" steam from other sources, then the incoming steam would tend to pump air into the upper downcomer and air accumulation might block incoming steam. However, the staff doesn't expect downcomer surface interactions to be uniform or necessarily pseudo steady-state around the circumference of the downcomer, and such simplistic arguments are weak.

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6 Initial RCS blowdown may leave the cross-over pipes empty and the SG secondary sides will remain pressurized. Initially, steam flowing through the hot legs will be superheated, with the amount of superheat diminishing with increasing time. Also typically, operators will at some time depressurize the SGs, eliminating the superheated steam.

Overall, the staff doesn't expect a large steam flow rate from the broken loop and judges this potential for transferring heat to incoming ECCS water to be small.

Steam flow from the upper plenum to the downcomer via the unbroken loops is also of concern. A large break LOCA may blow all water out of the cross-over pipes between the SGs and the reactor coolant pumps, thus establishing a steam flow path through all hot legs and SGs into the cold legs. Most or all of the steam from the broken cold leg will flow out the break and is of little concern if steam is also flowing through the other three legs. Steam in the other three cold legs will flow past the incoming ECCS water and into the upper downcomer on its path to the RV nozzle that is connected to the broken cold leg. This steam, and steam entering via the spray nozzles, will heat incoming ECCS water, thus challenging the licensee's 170 °F assumption. The licensee argues that this steam/water interaction will occur at the surface of the ECCS flow in the RCS cold legs and the top of the downcomer, and that this will lower the density of the heated water. It anticipates that a layer of hot water is likely to form near the steam that flows out the broken cold leg and is unlikely to descend to the bottom of the downcomer. The licensee provided no confirmatory information to support this argument and it failed to address interactions as the ECCS water enters the cold legs containing flowing steam. As stated at the end of the previous paragraph, the staff doesn't expect downcomer surface interactions to be uniform or necessarily pseudo steady-state around the circumference of the downcomer, and such simplistic arguments are weak. In addition, the staff expects significant interaction between incoming ECCS water and flowing steam prior to the ECCS water entering the downcomer.

The licensee did not assess its request with respect to risk. Although risk is not a "test" for meeting a regulation, the staff does consider risk in assessing the justification necessary to meet the regulations. In this case, the risk is assessed to be low because, as identified earlier in this safety evaluation, there are conservatisms that, if applied individually in a realistic analysis, would likely predict a hot leg injection initiation time of greater than 8.5 hours. In addition, the analyses are based on one operable ECCS system, consistent with the regulatory requirement to include the single worst failure when analyzing LOCAs. In most realistic considerations, the full complement of ECCS equipment would be operable. Operating two low pressure coolant injection pumps would significantly increase the fraction of steam heat that could be removed while remaining consistent with the licensee's assumed lower plenum temperature.

With this perspective, the licensee's contention is restated:

> If the lower plenum temperature is 170 °F, then the predicted maximum time for initiation of hot leg injection is 8.5 hours. The time-averaged RHR heat exchanger outlet temperature is approximately 150 °F. This 20 °F temperature difference is sufficient to condense 33 percent of the steam generated in the core. Given the physical processes, it is judged that steam and water mixing would be substantially less than 33 percent. As a result, the simplified assumption of a 170 °F lower plenum temperature as bounding is considered justified. Additional conservatism used in the modeling, in particular the core volume simplifications, provide additional assurance that the 8.5 hour time remains conservative for the Byron Station and Braidwood Station at uprated power operations.

The staff finds that this contention is not adequately supported for long-term operation because the licensee has only justified via an acceptable evaluation model that interaction with 33 percent of the steam can be accommodated when assuming a 170 °F lower plenum temperature.

On the basis the staff's evaluations, it believes that a realistic calculation would predict a switchover time greater than 8.5 hours. Further, the staff believes that removal of one or two of the many evaluation model conservatisms would provide a similar prediction. In addition, the staff judges that the licensee should be able to justify its 8.5 hour request via acceptable modifications to its evaluation model. Consequently, the staff will accept continued operation of the plants with the existing 8.5 hour switchover time for a period of 18 months from the date of this amendment and supporting Safety Evaluation. By letter dated April xx, 2001, the licensee stated that an analysis that justifies the time for switchover to hot leg recirculation will be submitted by June 1, 2002. The staff finds this to be acceptable.

3.2 Non-LOCA Transient Analysis

The licensee stated that the non-LOCA accident analysis methodology used to support the power uprate is the same methodology that is used for the current Byron and Braidwood licensing basis non-LOCA analyses with one exception. The exception is the use of a modified method which credits the effects of heat removal from the reactor coolant by the thick metal in the RCS during heatup portions of the feedwater line break accident. The licensee stated that this model change will lead to more realistic modeling of the feedwater transient and the staff agrees.

Where applicable, the non-LOCA analyses continue to employ the revised thermal design procedure (RTDP) methodology to determine the design limit DNBR value. The safety analysis limit DNBR was revised from 1.40 to 1.33.

The licensee also revised the over temperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoint values used in the safety analyses based on the new safety analysis DNBR limits and core thermal limits applicable for the uprated power conditions. With the exception of the f(ΔI) function setpoints for the OTΔT trip, the OTΔT and OPΔT trip setpoints remain unchanged. The power increase results in an increase in rod average linear power from 5.45 kW/ft to 5.73 kW/ft.

Thermal design flow (TDF) is increased from 358,800 gpm to 368,000 gpm as a result of reductions in the assumed maximum steam generator tube (SG) plugging levels (from 20 percent to 5 percent for the BWI SGs and from 24 percent uniform/30 percent peak to 10 percent uniform for the D5 SGs). A maximum 5 percent loop-to-loop flow asymmetry continues to be considered in the safety analysis consistent with the current licensing basis analyses. Corresponding to the increase in TDF, the minimum measured flow (MMF) used in conjunction with the RTDP DNBR methodology increased from 366,000 gpm to 380,900 gpm. Core bypass flow conditions remain consistent with those currently supporting thimble plug elimination and, as such, are not a change. The maximum reactor vessel average coolant temperature (T_{avg}) decreased from 588.4 °F to 588.0 °F. The minimum T_{avg} increased from 569.1 °F to 575.0 °F. Feedwater temperature at full power conditions increased from 440 °F to 446.6 °F. The feedwater temperature at hot zero power conditions remains at 100 °F.

Feedwater temperatures at part-power conditions increase proportionally with power between hot zero power and full power conditions.

The acceptance criteria for the anticipatory operational occurrences (AOOs) analyzed are that the calculated minimum DNBR remains greater than the safety limit, the peak RCS pressure remain less than the safety limit of 110 percent of design pressure (i.e., 2750 psia) and fuel centerline temperatures remain below the UO_2 melting point.

In determining the most limiting conditions for each event, the licensee considered both the BWI SGs (Unit 1) and the D5 SGs (Unit 2) in the analyses.

The results of the licensee's re-analyses for the spurious safety injection (SI) event indicated that the pressurizer safety valves (PSVs) will discharge liquid water for a time period of approximately 20 minutes. In order to confirm that the PSVs will discharge the necessary quantity of water and successfully reseat without sticking open, the staff requested additional information from the licensee regarding the qualification testing performed by the Electric Power Research Institute (EPRI) for the plant model PSVs for the applicable fluid inlet conditions for the spurious SI event. In a submittal dated January 31, 2001, the licensee provided the requested information as discussed below.

The licensee determined that relief of subcooled water was part of the EPRI testing of the Crosby PSVs (Reference: EPRI Report #NP-2770-LD, Volumes 1 and 6). Two water relief tests were performed at a water temperature as low as 635 °F (i.e., Test #926 with lowest temperature between 635 °F and 640 °F and Test #931 with lowest temperatures near 640 °F) and another performed at a water temperature of approximately 530 °F (i.e., Test #932). The results of the tests at 635 °F - 640 °F show stable valve operation. During the testing at 530 °F, the test valve experienced valve chatter that resulted in damage to the valve internals. However, as indicated in EPRI Report No. NP-2770-LD Volume 1, page S-6, in all cases, the safety valve closed in response to system depressurization.

The licensee has determined that the lowest water temperature predicted for the expected duration (i.e., 20 minutes) of the spurious SI transient at Byron Units 1 and 2 and Braidwood Units 1 and 2 is significantly higher (i.e., 590 °F) than the lowest temperature (i.e., 530 °F) for the EPRI tests. The licensee states that, although stable valve operation cannot be assured, any valve damage would be expected to be less than the damage experienced during the EPRI testing and that the PSVs will close upon system depressurization. The licensee concludes that the spurious SI event does not progress into a stuck open PSV LOCA event and that all three PSVs may lift in response to the event, but they will reclose. The licensee states that the resulting leakage from up to three PSVs is bounded by flow through one fully open PSV, which is an analyzed event.

The duration of the spurious SI event is no more than 20 minutes from the initial SI signal to the time when system pressure is restored to below the PSV lift setpoint. The inadvertent SI event is terminated by operator action. The licensee's analyses show that during this 20 minute time frame, a PSV will cycle a number of times (i.e., approximately 20) with the valve being open for 5-8 seconds per cycle. The licensee states that only one PSV is required to mitigate the pressure transient, and that even though the three PSVs are set to lift at the same pressure, from a statistical standpoint, one valve would lift earlier than the other two. This would result in no more than one valve being challenged at a time.

The staff has reviewed the licensee's evaluation of the performance of the plant PSVs for the liquid water conditions during a spurious SI event. The staff finds that the EPRI tests adequately demonstrate the performance of the valves for the expected water temperature conditions and that there is reasonable assurance that the valves will adequately reseat following the spurious SI event. A review of the above stated EPRI test data indicates that the PSVs may chatter for the expected fluid inlet temperature but that the resulting PSV seat leakage following the liquid discharge would be less than the discharge from one stuck-open PSV, which is an analyzed event. Therefore, the staff finds the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable.

The feedwater line break (FWLB) analysis also results in liquid water discharge through the PSVs and has been previously evaluated by the licensee in the current licensing basis. The staff has reviewed the information provided by the licensee regarding the change to the temperature of the liquid discharge through the PSVs as a result of power up-rate. The temperature of the liquid discharge for the FWLB is very similar to the current licensing basis conditions, and the performance of the PSVs would also be similar. Therefore, the performance of the PSVs for the FWLB event is acceptable.

3.2.1 Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition

The licensee analyzed the uncontrolled RCCA bank withdrawal from a subcritical condition event using methods that the staff has previously approved to ensure that the core and the RCS are not adversely affected by the proposed power uprate. The results of the licensee's analysis indicate a minimum DNBR greater than the safety analysis limit of 1.33 and maximum fuel temperatures much less than those required for fuel melting (4800 °F). Therefore, no fuel melting or clad damage is predicted as a result of this transient at uprated conditions. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

The licensee analyzed the uncontrolled RCCA bank withdrawal from power conditions event using methods that the staff has previously approved to ensure that the core and the RCS are not adversely affected by the proposed power uprate. The results of the licensee's analysis show that the minimum value of DNBR is always larger than the safety analysis limit of 1.33, and the RCS and main steam system are maintained below 110 percent of their design pressures. Thus the event does not adversely affect the core, RCS, or main steam system (MSS), and is protected by the high neutron flux and OTΔT trips over the entire range of possible reactivity insertion rates. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of the analysis met the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.3 Rod Cluster Control Assembly Misoperation

Misoperation events include a dropped RCCA or dropped bank, RCCA misalignment, and single RCCA withdrawal. For the drop and misalignment events, DNB does not occur.

Because of the low probability of the combination of conditions required to cause a single RCCA withdrawal, it is considered an infrequent fault with a fuel damage limit set at 5 percent of the total fuel rods. The results of the licensee's analysis for a single RCCA withdrawal event show that the number of fuel rods experiencing a DNBR below the safety analysis limit is less than 5 percent of the total fuel rods in the core. Therefore, the applicable acceptance criteria for these events continue to be met at uprated power conditions. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of the analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.4 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (Uncontrolled Boron Dilution)

The licensee analyzed this event to ensure that there is sufficient time for mitigation of an inadvertent boron dilution prior to complete loss of shutdown margin. Inadvertent dilution during refueling (Mode 6) is precluded through administrative control of valves in the possible dilution flow paths. By amendments 117 for Byron Units 1 and 2, and 111 for Braidwood Units 1 and 2, issued on April 6, 2001, the NRC approved the removal of the boron dilution protection system (BDPS) for the plants. In its place, the units will rely on an alternative system of new alarms, indicators, procedures, and controls, and the operators to mitigate a boron dilution event, should it occur in Modes 3, 4, or 5. The staff determined that there is reasonable assurance that the Byron and Braidwood plant operators will perform the required manual actions necessary to mitigate both slow and fast dilution events.

For Modes 1 and 2, an inadvertent boron dilution would be terminated by plant operator actions after being alerted to the dilution event by a reactor trip, on source range, neutron flux high, power range neutron flux high, or on OTΔT, or after being alerted by the low and low-low control rod insertion limits.

Based on the above, the staff concludes that Byron and Braidwood are adequately protected for boron dilution events.

3.2.5 Partial Loss of Forced Reactor Coolant System Flow

The licensee analyzed a partial loss of reactor coolant flow event (which involves the loss of two reactor coolant pumps (RCPs) with four loops in operation) at power uprate conditions using methods that the staff has previously approved to confirm that the conclusions in the current analysis remain valid. The results of the licensee's analysis show that the minimum DNBR is greater than the safety analysis limit of 1.33 and the peak primary and secondary system pressures are below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.6 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The current TSs at Byron and Braidwood preclude power operation with an inactive loop. Therefore, this event is not analyzed.

3.2.7 Loss of External Electrical Load and/or Turbine Trip

The licensee analyzed the loss of external electrical load and/or turbine trip event at the power uprated conditions using methods that the staff has previously approved. In the minimum DNBR case, the pressurizer power operated relief valves (PORVs) and pressurizer spray portion of the automatic pressure control system are assumed to function during the transient since these features will limit the RCS pressure increase, which is conservative to DNBR calculation. The results of the licensee's analysis show that the minimum DNBR remains above the safety limit of 1.33. In the peak pressure case, the PORVs and pressurizer spray are not assumed to function but the pressurizer and steam generator safety valves are actuated. The results of the licensee's analysis show that the peak primary and secondary systems are maintained below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.8 Loss of Normal Feedwater

The licensee analyzed the loss of normal feedwater event at the power uprated conditions using methods that the staff has previously approved. The DNB transient for this event is bounded by the complete loss of forced reactor coolant flow event which demonstrated that the minimum DNBR is greater than the safety limit value. The results of the licensee's analysis show that pressurizer does not reach a water solid condition. In the licensee's analysis, the PORVs and pressurizer sprays are assumed to be operable to maximize the pressurizer water volume. The pressure transient following a loss of normal feedwater event is bounded by the more pressure limiting loss of load/turbine trip event which is discussed in Section 3.2.7 of this SE. The analysis of the pressure bounding loss of load/turbine trip event demonstrates that the peak primary and secondary system pressures are maintained below 110 percent of their respective design pressures which assumes the PORVs and pressurizer sprays are unavailable. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis met the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.9 Loss of Non-Emergency AC Power to the Plant Auxiliaries

The licensee analyzed the loss of non-emergency AC power to the plant auxiliaries event for the power uprate using methods that the staff has previously approved. The DNB transient for this event is bounded by the complete loss of forced reactor coolant flow event which demonstrated that the minimum DNBR is greater than the safety limit value. The results of the licensee's analysis show that pressurizer does not reach a water solid condition and that the peak primary and secondary pressure remain below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.10 Excessive Heat Removal Due to a Feedwater System Malfunction

The licensee analyzed both of the most limiting excessive feedwater flow case and the most limiting feedwater temperature reduction case at the power uprated conditions using methods that staff has previously approved. The results of these analyses show that the minimum DNBRs are greater than the safety analysis limit of 1.33. Since these events are primarily cooldown events, over pressurization limits for the primary and secondary systems are not challenged for these events. The staff has reviewed the assumptions and the results of the licensee's analyses and concluded that the assumptions used in these analyses are conservative and the results of these analyses met the acceptance criteria for these events. Therefore, the staff finds the licensee's analyses acceptable.

3.2.11 Accidental Depressurization of RCS

This event could occur due to inadvertent opening of a pressurizer relief or safety valve. Since the pressurizer safety valve has larger relieving capacity, an inadvertent opening of a safety valve is more limiting. The licensee analyzed this case at power uprate conditions using methods that the staff has previously approved. The results of this licensee's bounding analysis show that the minimum DNBR is greater than the safety analysis limit of 1.33 and the peak primary and secondary pressures remain below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.12 Inadvertent Operation of Emergency Core Cooling System During Power Operation

The licensee analyzed this event at power uprate conditions using methods that the staff has previously approved. The results of the licensee's analyses show that the pressurizer will become water solid during this event. The staff acceptability regarding the potential liquid relief through the pressurizer safety valves is discussed in Section 3.2 of this SE. The results of the licensee's analysis show that the minimum DNBR never falls below the initial value which is greater than the safety analysis limit of 1.33 and the peak primary and secondary pressures remain below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.13 Inadvertent Loading of a Fuel Assembly into an Improper Position

The licensee analyzed this event to verify that if a loading error exists during operation at the uprated power, the resulting power distribution effects would either be readily detected by the incore moveable detector system or cause a sufficiently small perturbation to permit continued reactor operation. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.14 Complete Loss of Reactor Coolant Flow

The licensee analyzed two complete loss of forced reactor coolant flow cases at power uprate conditions using methods that the staff had previously approved. They are: 1) complete loss of power to all RCPs and 2) RCP power supply frequency decay. The licensee's analysis of case 2 provides more limiting results due to its delayed reactor trip on under-frequency trip. The results of this bounding analysis show that the minimum DNBR is greater than the safety analysis limit of 1.33 and the peak primary and secondary pressures remain below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.15 Single Rod Cluster Control Assembly Withdrawal at Full Power

This accident analysis is addressed in Section 3.2.3 of this SE.

3.2.16 Excessive Load Increase Incident, Accidental Depressurization of Main Steam System, Minor Secondary System Pipe Breaks, and Rupture of a Main Steamline

3.2.16.1 Excessive Load Increase Incident

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The licensee analyzed scenarios that include a combination of manual or automatic rod control associated with minimum and maximum reactivity feedback at the power uprated conditions using methods that the staff has previously approved. The results of these analyses show that the minimum DNBRs are greater than the safety analysis limit of 1.33 and the peak primary and secondary pressures remain below 110 percent of their respective design pressures. The staff has reviewed the assumptions and the results of the licensee's analyses and concluded that the assumptions used in these analyses are conservative and the results of these analyses meet the acceptance criteria for these events. Therefore, the staff finds the licensee's analyses acceptable.

3.2.16.2 Accidental Depressurization of Main Steam System

The accident is addressed in Section 3.2.16.5 of this SE.

3.2.16.3 Minor Secondary System Pipe Breaks

This accident is addressed in Sections 3.2.16.5 and 3.2.20.1 of this SE.

3.2.16.4 Rupture of a Main Steamline

This accident is addressed in Section 3.2.20.1 of this SE.

3.2.16.5 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve creates a depressurization of the secondary system with an effective opening size within the spectrum of break sizes analyzed in the main steam line break event described in Section 3.20.1 and 3.20.2 of this SE. In responses to the staff request for additional information, the licensee has stated that the calculated minimum DNBR is 1.838 for the bounding steam line break accident at power uprated conditions. Therefore, the expected minimum DNBR during an inadvertent opening of a steam generator relief of safety valve will be greater than the safety analysis limit of 1.33. The allowable peak primary and secondary system pressure will not be challenged since this is a cooldown event. The staff has reviewed the assumptions and the results of the licensee's analyses and concluded that the assumptions used in these analyses are conservative and the results of these analyses meet the acceptance criteria for these events. Therefore, the staff finds the licensee's analyses acceptable.

3.2.17 Feedwater System Pipe Break

The licensee has analyzed the feedwater system pipe break accident at the power uprate conditions. The methodology used for the licensee's analysis was modified from that in the current analysis to credit the effects of heat removal from the reactor coolant by the thick metal in the RCS during the heatup portion of the event. This is a more realistic modeling of the transient. Both of the new and current analyses show that the pressurizer will become water solid during this event. The staff acceptability regarding the potential liquid relief through the pressurizer safety valves is discussed in Section 3.2 of this report. Depending on the conditions of the break, the feedwater line break could cause either an RCS cooldown or an RCS heatup. The effect of RCS cooldown resulting from a secondary system pipe break is bounded by the main steam line break analyses since steam blowdown will result in a more excessive cooldown than water blowdown through a rupture in the main feedwater line. The primary and secondary system peak system pressures are bounded by the more limiting loss of load/turbine trip event discussed in Section 3.2.7 of this SE. The analysis for the bounding loss of load/turbine trip event demonstrates that the peak primary and secondary pressures are maintained below 110 percent of respective design pressures. The results of the licensee's analysis shows that the assumed auxiliary feedwater system capacity is adequate to remove core decay heat and to prevent uncovering the reactor core for the postulated feedwater line break at the power uprate conditions. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.18 Single Reactor Coolant Pump Locked Rotor/Shaft Break

The licensee analyzed the single reactor coolant pump lock rotor/shaft break accident at power uprate conditions using methods that the staff has previously approved. The results of the licensee's analysis show that the peak primary and secondary pressures remain within 110 percent of their respective design pressures. The maximum clad temperature is 1954 °F. Although DNB occurs, the number of fuel rods in DNB is less than that assumed in the radiological assessment for this event. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are

conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable.

3.2.19 Rod Cluster Control Assembly (RCCA) Ejection

The licensee analyzed this accident at power uprate conditions. The results of the RCCA ejection accident indicate that the average fuel enthalpy at the hot spot remains well below 280 cal/gm and therefore, there is no danger of sudden fuel dispersal into the coolant. DNB is predicted to occur in less than 10 percent of the core, thus limiting fission product release. Peak RCS pressure does not exceed required stress limits and, thus, there is no danger of further consequential damage to the RCS. Therefore, the consequences of an RCCA ejection analysis at uprated power remain acceptable. The staff has reviewed the results of the licensee's analysis and finds it acceptable.

3.2.20 Steam System Piping Failure

3.2.20.1 Steam System Piping Failure at Zero Power

The licensee analyzed the steam system piping failure at zero power event at power uprate conditions using methods that the staff has previously approved. The rupture of a major steam line is the most limiting cooldown transient. The accident is analyzed with no decay heat to optimize the cooldown rate. The licensee's analysis conservatively assumed the most reactive RCCA stuck in its fully withdrawn position and assumed a single failure in the engineering safety features. The licensee performed the analysis both with and without offsite power available. The licensee determined that the case with off-site power available is the limiting case. The steam system piping failure event is classified as an event of limiting faults (condition IV event under Westinghouse classification) which allows some fuel failures. However, the results of the licensee's analysis of the bounding case show that the minimum DNBR is greater than the safety limit of 1.33, therefore, the licensee's analyses would predict that no fuel failure occur. The licensee's analyses also demonstrated that the calculated peak primary and secondary system pressure did not challenge the allowable peak primary and secondary system pressures. The staff has reviewed the assumptions and the results of the licensee's analyses, and concluded that the assumptions used in these analyses are conservative and the results of these analyses meet the acceptance criteria for these events. Therefore, the staff finds the licensee's analyses acceptable.

3.2.20.2 Steam System Piping Failure at Full Power

The licensee's analysis of a main steam line break at zero power represents the limiting condition with respect to core performance during the event. Also, the licensee's analysis demonstrated core protection in coping with the situation associated with return to power after reactor trip. The purpose of the analysis of a main steam line break at full power is to demonstrate that core protection is maintained prior to and immediately following reactor trip. The steam system failure at full power event was analyzed at power uprate conditions using methods that the staff has previously approved. Cases are analyzed with various break sizes. This steam system failure at full power event is classified as a event of limiting faults (condition IV event under Westinghouse classification) which allows some fuel failures. However, the results of the analysis of the bounding case show that the minimum DNBR is greater than the safety limit of 1.33 and, therefore, the licensee's analyses predicted that no

fuel failure would occur. The licensee's analyses also demonstrated that the calculated peak primary and secondary system pressure do not challenge the allowable peak primary and secondary system pressures. The staff has reviewed the assumptions and the results of the licensee's analyses and concluded that the assumptions used in these analyses are conservative and the results of these analyses meet the acceptance criteria for these events. Therefore, the staff finds the licensee's analyses acceptable.

3.2.21 Steam Generator Tube Rupture (SGTR)

The licensee analyzed the steam generator tube rupture accident at the power uprate conditions using methods that the staff has previously approved. Two separate analysis were performed to cover different steam generator design between Units 1 and Units 2 at Byron and Braidwood Stations. The results of both analyses show that there is sufficient margin to overfill in the steam generators prior to the operators taking control of the auxiliary feedwater flow rate. The staff has reviewed the assumptions and the results of the licensee's analysis and concluded that the assumptions used in this analysis are conservative and the results of this analysis meet the acceptance criteria for this event. Therefore, the staff finds the licensee's analysis acceptable. The radioactive steam released to environment during the event were generated from the analyses for assessment of radiological consequences addressed in Section 3.5.3 of this SE.

3.3 Containment Integrity Analyses

The licensee performed containment integrity analyses at uprated power to ensure that the maximum pressure inside the containment would remain below the containment building design pressure of 50 psig if a design bases loss of cooling accident (LOCA) or main steam line break (MSLB) inside containment would occur during plant operation. The analyses also established the pressure and temperature conditions for environmental qualification and operation of safety related equipment located inside the containment. The LOCA peak pressure was also used as a basis for the containment leak rate test pressure to ensure that dose limits would be met in the event of a release of radioactive material to containment.

The licensee indicated that the containment functional analyses included the assumption of the most limiting single active failure and the availability or unavailability of offsite power, depending on which resulted in the highest containment temperature and pressure. Bounding initial temperatures and pressures for analyses were selected to envelop the limiting conditions for operation. The containment integrity analysis is presented in attachment E of the July 5, 2000, submittal.

3.3.1 LOCA Containment Analyses

The licensee performed analyses to determine the containment pressure and temperature response during postulated LOCAs using mass and energy releases which incorporate revised design parameters corresponding to 3586.6 MWt plus a 2 percent allowance for calorimetric error with updated computer modeling. As in the current Byron/Braidwood Updated Final Safety Analysis Report (UFSAR), the postulated LOCA analyses were performed for the double-ended hot leg (DEHL) guillotine break and the double-ended pump suction (DEPS) break of reactor coolant pipe. It has been determined that the DEHL break results in the most

limiting pressure during the blowdown phase and that the DEPS break yields the highest energy flow rates during the post-blowdown period.

The licensee indicated that the mass and energy releases in the containment were calculated for power uprate using Westinghouse Topical Report WCAP-10325-P-A. In this uprate analyses, the 1979 ANS 5.1 decay heat model with 2 sigma uncertainty factor was used. Westinghouse Topical Report WCAP-8264-P-A for mass and energy release calculations was used for the current design bases analyses. Separate analyses were performed for Byron and Braidwood Units 1 which have the BWI steam generators, and Byron and Braidwood Units 2 which have Westinghouse D5 steam generators. The updated Westinghouse Topical Report WCAP-10325-P-A computer code uses the same methodology and assumptions (except for the Byron and Braidwood specific data) that have been utilized and approved on many plant-specific dockets for Westinghouse PWRs. The staff finds the use of Topical Report WCAP-10325 for LOCA mass and energy release calculations acceptable.

The mass and energy releases calculated by the above analyses were utilized for the power uprate containment pressure and temperature response analyses using the Westinghouse computer code COCO. The current Byron and Braidwood containment temperature and pressure analyses were also performed using the COCO computer code. This code has been used and found acceptable for many dry containment plants and is acceptable for use at Byron and Braidwood.

For Byron and Braidwood Units 1, the analyses for the power uprate calculated a containment peak pressure of 42.8 psig and a peak temperature of 264.5 °F for the DEHL pipe break. For Byron and Braidwood Units 2, the uprating analyses calculated a peak pressure of 38.4 psig and a peak temperature of 257.6 °F for the DEHL pipe break. The LOCA analysis also showed that the containment pressure for all Byron and Braidwood units was reduced to less than 50 percent of the peak calculated pressure within 24 hours. The current peak containment pressure calculated for Byron and Braidwood Units 1, was 47.8 psig and for Byron and Braidwood Units 2 was 44.4 psig. The uprate calculated LOCA peak pressure and temperature for both Byron and Braidwood Units 1, and Byron and Braidwood Units 2, remains below the containment design pressure of 50 psig and design temperature of 280 °F. Based on the above, the staff finds that the power uprate will not impact containment integrity for a design bases LOCA event.

The licensee has proposed to revise the Byron and Braidwood TSs for containment leak rate testing based on the calculated uprate peak LOCA containment pressure of 42.8 psig for Units 1 and 38.4 psig for Units 2. The staff finds the proposed TS change acceptable.

3.3.2 Main Steamline Break Containment Integrity Analysis

The licensee has performed analyses to determine the containment pressure and temperature response during postulated main steamline breaks (MSLBs) inside containment for limiting conditions for operation at uprated power. As in the current licensing basis UFSAR, the uprated analyses were evaluated for power levels and a spectrum of break sizes similar to that in the current UFSAR. The MSLB mass and energy releases at the uprate power were calculated using the Westinghouse LOFTRAN computer code. The same code was used in the current licensing basis analysis. The staff finds the use of LOFTRAN computer code for calculating MSLB mass and energy releases is acceptable.

The mass and energy releases calculated from the above analyses were utilized for uprated containment pressure and temperature response analyses using the Westinghouse computer code COCO. The current Byron and Braidwood MSLB containment temperature and pressure analyses were also performed using the COCO computer code. The staff has found the use of this code acceptable.

For the Byron and Braidwood, Units 1, the MSLB uprating analyses calculated a peak containment pressure of 39.3 psig and a peak containment temperature of 333 °F at 102 percent of uprate power level. For the Byron and Braidwood Units 2, the uprating analyses calculated a peak containment pressure of 38.3 psig and a peak containment temperature of 331 °F at 102 percent of uprate power level. The peak containment temperatures at current power level were also 333 °F for Byron and Braidwood, Units 1 and 331 °F for Byron/ Braidwood Units 2. The peak containment pressure at uprated conditions remains below the containment design pressure of 50 psig. The licensee indicated that the time duration of the containment peak air temperature is very short and that the containment structure temperature will remain below the containment design temperature of 280 °F. Also the updated calculated pressure and temperature curves for LOCA and MSLB cases will remain bounded by the curves used for equipment qualifications.

Based on the above evaluation, the staff finds the proposed change for power uprate will not affect the containment design because the calculated peak containment pressure remains below the containment design pressure of 50 psig and the containment structure will remain below its design temperature of 280 °F. Therefore, the staff finds that the power uprate will not impact containment integrity for a design bases MSLB event.

3.3.3 Short-term Subcompartment Analysis

The licensee indicated that the short-term LOCA-related mass and energy releases that support subcompartment analyses were reviewed to assess the effects associated with power uprate. The subcompartments evaluated include the steam generator compartment, reactor cavity region, and pressurizer compartment. The Byron and Braidwood Units 1 and 2 are approved for leak before break (LBB) which eliminates the dynamic effects of postulated primary loop pipe ruptures from the design basis. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg break for the steam generator compartments, and a 150 in^2 reactor vessel inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since these units are approved for LBB, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with the power uprate conditions. The current licensing basis subcompartment analyses that consider breaks in the primary loop reactor coolant system piping (steam generator subcompartment and reactor cavity region), therefore, remain bounding.

The short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures. For the pressurizer compartment, the licensee indicated that the critical mass flux correlation was used to conservatively estimate the impact of changes in RCS temperatures on the short-term releases. The evaluation showed that the releases based on

the power uprate conditions were bounded by the releases documented in the Byron and Braidwood Stations' UFSAR and that the short-term pressurizer subcompartment loading analyses will remain acceptable. Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications from similar PWR plants, the staff agrees with the licencee's conclusion that plant operations at the proposed uprated power level will have an insignificant or no impact on the short-term subcompartment analysis.

3.4 Additional Design Basis and Programmatic Evaluations

3.4.1 Containment Post-LOCA Combustible Gas Control

The licensee indicated that the effect of power uprate was reviewed for post-LOCA hydrogen production resulting from the Zirconium-water reaction, corrosion of construction materials in the containment, and radiolysis of aqueous solution in the core and in the sump, and for the capability of the combustible gas control system to maintain hydrogen concentration from exceeding the lower flammable limit of 4.0 percent by volume inside the containment. The hydrogen produced both at the current power level and at the uprated power level were calculated according to the method described in UFSAR Section 6.2.5 "Combustible Gas Control in Containment." The calculation to determine the hydrogen concentration was revised to reflect the power uprate conditions. The revised input included the post-LOCA containment temperature curve which affects the corrosion of metals in the containment, revised decay heat which affects the radiolysis of the coolant, and revised core wide oxidation of the zirconium fuel cladding and the reactor coolant from 0.82 percent for the pre-uprate to 1 percent for the uprate.

The licensee indicated that although the impact of the power uprate on the combustible gas control system is an increase in the maximum hydrogen concentration in containment post LOCA, the 4 percent hydrogen concentration limit is not exceeded. The existing design of the combustible gas control is able to maintain the hydrogen concentration below 4 percent provided a single 65 scfm hydrogen recombiner is operating 20 hours post accident and run continuously thereafter. The power uprate design is also able to maintain the hydrogen concentration below 4 percent provided a single 65 scfm hydrogen recombiner is operating 20 hours post accident and run continuously thereafter. The maximum hydrogen concentration in the containment is calculated to reach 3.78 percent after 11.6 days at the current power level and 3.93 percent after 12.7 days at uprated power with one recombiner operating 20 hours post accident.

The licensee also indicated that with no recombiner in operation, and assuming containment purge start at 5 days post-LOCA and run continuously, hydrogen concentration is calculated to remain below 4 percent after power uprate. Therefore, the licensee determined that the power uprate does not impact the post LOCA combustible gas control system's ability to maintain the hydrogen concentration below 4 percent.

Based on the review of the licensee's rationale and the experience gained from the staff's review of power uprate applications from similar PWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the post LOCA combustible gas control and the system will continue to perform its design function at the uprate power level.

3.4.2 Compliance with 10 CFR Part 50, Appendix R

The licensee indicated that the compliance with the Fire Protection (Appendix R) Program will not be affected because the power uprate evaluation did not identify changes to design or operating conditions that will adversely impact the Appendix R post-fire safe shutdown capability. Based on the experience gained from the staff's review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the compliance with the licensee's 10 CFR Part 50, Appendix R program

3.4.3 Station Blackout (SBO)

The licensee evaluated the impact of plant operations at the proposed uprated power level on systems required to cope with SBO events. The licensee stated that current design basis temperature profiles in areas housing SBO required equipment remain bounding for an SBO event.

The staff was concerned that the plant response and scoping capabilities for SBO might be affected by operation at the uprated power level due to the increase in operating temperature of the primary coolant system and increase in decay heat. In a request for additional information, the staff requested the licensee to discuss and verify that the assumptions for the existing SBO analysis are valid for the power uprate conditions, particularly the heatup analysis, equipment operability, and battery capacity. In response to the staff's request, the licensee stated that in evaluating the systems impacted by the uprate, it had not identified any changes to assumptions, design, or operating conditions that would adversely affect the ability to provide safe shutdown for an SBO. In addition, the power uprate will not create any additional electrical demands or require any equipment modifications that would affect the plant heatup analysis or increase battery loading. However, power uprate will increase decay heat load during the coping period. The increased decay heat will require an increase in the total volume of water, and this will be supplied by the auxiliary feedwater system during the coping period. However, sufficient useable inventory in the condensate storage tank (CST) is available to satisfy AFW requirements for plant cooldown.

Since the plant response and scoping capabilities for SBO will not be affected significantly by operation at the uprated power level, the staff concludes that the power uprate conditions would not significantly affect the previous SBO analysis.

3.4.4 Safety-Related Motor-Operated Valves (MOVs)

In its January 31, 2001, response to the staff's request for additional information, the licensee stated that the Braidwood station and Byron station NSSS and BOP safety-related valves (i.e., main steam safety valves, power operated relief valves, and main steam isolation valves) were capable of meeting their performance requirements for the power uprate conditions and, therefore, are acceptable. The licensee confirmed its conclusion by verifying that the uprated system operating temperature, pressure and flow were within the acceptance criteria of the associated equipment specification.

The licensee also indicated that the impact of increased parameters on the design basis pressures used in the Generic Letter (GL) 89-10 Safety-Related "Motor-Operated Valve" (MOV)

program was evaluated. The increased flow requirement in some safety related systems due to power uprate, will increase the differential pressures across the associated MOVs. As a result, the licensee concluded that the power uprate has no adverse impact on the Braidwood and Byron Generic Letter 89-10 MOV program. As stated by the licensee, this is because Braidwood and Byron station evaluations in response to Generic Letter 89-10 MOV program were conservatively based on pump shutoff head, relief and safety valve setpoints plus accumulation, containment design pressure, and interlock setpoints which are not changed as a result of power uprate.

In addition, the licensee indicated that the revised post-accident temperature and pressure conditions for systems and components that are subject to pressure locking and thermal binding were not impacted, therefore, power uprate does not impact the GL 95-07 evaluations.

3.5 Radiological Analysis

To demonstrate that the Byron and Braidwood engineered safety features (ESFs) designed to mitigate the radiological consequences will remain adequate at uprated power level of 3586.6 MWt, the licensee reevaluated the offsite and control room radiological consequences for the following postulated design-basis accidents (DBAs) at a power level of 3658.3 MWt (102 percent of requested uprated power level of 3586.6 MWt):

Main Steamline Break
Locked Reactor Coolant Pump (RCP) Rotor
Locked RCP Rotor with Power-Operated Relief Valve (PORV) Failure
Rod Ejection
Small Line Break
Steam Generator Tube Rupture
Large-Break Loss-of-Coolant Accident (LOCA)
Small-Break LOCA
Fuel Handling Accident
Gas Decay Tank Rupture

The licensee submitted the results of its offsite and control room dose calculations. In addition, the licensee provided the major assumptions and parameters used in its dose calculations. As documented in the submittals, the licensee has determined that the existing ESF systems at Byron and Braidwood will still provide assurance that the radiological consequences of the postulated DBAs at the exclusion area boundary (EAB), in the low-population zone (LPZ), and in the control room are within the radiation dose acceptance criteria specified in the SRP and the dose limits proved in 10 CFR 100.

The staff has reviewed the licensee's analysis and has performed an independent confirmatory radiological consequence dose calculation for the following 6 bounding DBAs:

Large-Break Loss-of-Coolant Accident (LOCA)
Main Steamline Break
Steam Generator Tube Rupture
Fuel Handling Accident
Locked RCP Rotor with a Steam Generator PORV Failure
Rod Ejection

The results of the staff's independent radiological consequence calculations are given in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff are listed in Tables 3 through 14. The staff did not perform independent dose calculations for the small-break LOCA and the small-line break accident since the radiological consequences of these accidents at Byron and Braidwood stations are bounded by that of the large-break LOCA. The radiological consequences of the locked RCP rotor accident is also bounded by that of the accident with a steam generator PORV failure. The staff also did not perform an independent dose calculation for gas decay tank rupture because the quantity of radioactivity in each gas decay tank is limited by Byron and Braidwood TS 5.12 and the licensee did not request to change the limits for this TS.

In addition, the licensee requested to amend the definition of Dose Equivalent Iodine-131 in the Byron and Braidwood Technical Specification Section 1.1, "Definition." The current definition defines Dose Equivalent Iodine-131 as follows:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131(microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133 I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for power and Test Reactor Sites."

The requested amendment would add two following references to this definition: (1) Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, 1977, and (2) ICRP 30, "Limits for Intakes of Radionuclides by Workers," Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

The amended definition would then read as follows:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131(microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133 I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, 1977, or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity.

The International Commission on Radiation Protection Publication 30 (ICRP 30) incorporates the considerable advances in the state of knowledge of radionuclide dosimetry and biological transport in humans achieved in the past few decades and the NRC embraced it and adopted its values into the revision of Part 20, "Standards for Protection Against Radiation," in 1994. Therefore, the staff finds that this requested amendment to the definition is acceptable.

The following sections provide the staff's assessment of the potential consequences of the six postulated accidents.

3.5.1 Accidents Analyzed

3.5.1.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA using Technical Information Document (TID)-14844 source term is provided in Byron and Braidwood UFSAR Section 15.6.5. The licensee reevaluated the offsite and control room radiological consequences of the postulated LOCA at an uprated power level of 3658.3 MWt. The staff has reviewed the licensee's analysis and performed an independent confirmatory dose calculation for the following two potential fission product release pathways after the postulated LOCA:

(1) containment leakage
(2) post-LOCA leakage from ESF systems outside containment.

The current maximum allowable primary containment leakage rate (L_a), is 0.1 percent of containment air weight per day. The staff used this leak-rate for the first 24 hours of the accident and 0.05 percent of containment air weight per day for the remaining duration of the accident (30 days). Only fission product removal in the containment atmosphere is achieved by the containment spray system (CSS) other than initial plateout in the containment assumed in the source term. The CSS is an ESF system and is designed to provide containment cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 2950 gpm.

The licensee calculated the elemental iodine removal rate by the CSS using the methodologies provided in SRP Section 6.5.2 and determined that elemental iodine removal rate to be well above the upper limit specified in the SRP. Therefore, the licensee used an elemental iodine removal rate of 20 per hour specified in the SRP as an upper limit. The licensee also calculated a removal rate of iodine in particulate form using the methodologies provided in SRP Section 6.5.2 and determined the rate to be 6 per hour. The staff finds these iodine removal rates determined by the licensee are acceptable. The licensee assumed removal of elemental iodine from the containment atmosphere only during spray injection period (from 0.025 hours to 0.373 hours following the accident) and determined that the decontamination factors (DFs) 100 and 50 referenced in the SRP for elemental iodine and iodine in particulate form respectively, are not reached during this spray injection period.

The licensee modeled the containment atmosphere as two discrete nodes representing sprayed and unsprayed regions and assumed these nodes are mixed by the reactor containment fan cooler (RCFC) system fans. The RCFC system is an ESF system and is designed to remove energy released in the containment following a postulated LOCA (along with the emergency core cooling system and the containment spray system). The RCFC system is a redundant system consisting two 100 percent trains. Each train is powered from a separate redundant essential bus and has a capacity of 1.18E+6 cfm air flow. The staff assumed that only one RCFC system train will be operational with a total air mixing flow rate of 1.06E+6 cfm (90 percent of fan capacity) in the containment following the postulated LOCA.

This represents a mixing rate of approximately 12 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere.

Any leakage water from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The licensee assumed this leakage to be less than 7820 cc/hour, which is twice the leakage value of 3910 cc/hour assumed in the Byron and Braidwood UFSAR, and that this leakage would begin at the time of the postulated LOCA and continue throughout the entire duration of the accident (30 days). The staff finds the leakage value assumed by the licensee to be acceptable. The licensee further assumed that ten percent of all forms of iodine contained in the leakage will be released (consistent with guidelines provided in SRP Section 15.6.5) to the environment through auxiliary building filtration system (ABFS) which is designed as an ESF system. The staff assumed 1 percent of the ABFS flow will bypass the charcoal adsorber in the ABFS.

The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria. The resulting radiological consequence analyses performed by the staff for the EAB, the LPZ, and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used for the postulated LOCA dose calculations by the staff are provided in Table 3. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood stations operating at the uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed the dose guidelines provided in 10 CFR 100 and the control room dose acceptance criteria specified in GDC 19.

3.5.1.2 Main Steamline Break Outside Containment (MSLB)

The licensee has reevaluated the radiological consequences of a postulated MSLB accident occurring outside containment and upstream of the main steam isolation valves at an uprated power level of 3658.3 MWt. The licensee analyzed this postulated accident using 0.5 gpm of primary-to-secondary leakage through the faulted steam generator and 0.218 gpm through each of the intact steam generators. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

The staff performed an independent radiological consequence calculation for two cases. For Case 1, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the MSLB accident. Before the accident, the reactor was assumed to be operating at its TS equilibrium limit of 1.0 µCi/gm dose equivalent iodine-131 (DEI-131) in the primary coolant. The iodine spike generated during the accident was assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing concentration in the primary coolant during the course of the accident. For Case 2, the staff assumed that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous TS limit of 60 µCi/gm DEI-131. For both cases, the staff

assumed that all fission products in the entire mass of secondary water in the faulted steam generator (167,000 lbs) was released to the environment directly with no iodine partition.

The resulting radiological consequence analyses for the EAB, the LPZ, and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff for the main steam line break accident are provided in Table 4. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood stations operating at the uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated main steamline break accident occurring outside containment will not exceed the dose acceptance criteria specified in SRP Section 15.1.5 and dose guidelines set forth in 10 CFR 100, and the control room dose acceptance criteria specified in GDC 19.

3.5.1.3 <u>Steam Generator Tube Rupture Accident</u>

The licensee has reevaluated the radiological consequences of a postulated steam generator tube rupture accident at an uprated power level of 3658.3 MWt and provided a radiological consequence analysis. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

To verify the licensee's assessments, the staff performed independent radiological consequence calculations for two scenarios for the steam generator tube rupture accident as the staff did for the steamline break accident above. For Case 1, the staff assumed that a temporary increase in the primary coolant iodine spike occurred as a result of the power/pressure transient caused by the steam generator tube rupture. Before the postulated accident, the Byron and Braidwood stations were assumed to be operating at their TS equilibrium iodine concentration limit of 1.0 µCi/gm DEI-131 in the primary coolant. The iodine spike generated during the accident was assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate resulted in an increasing iodine concentration in the primary coolant during the course of the accident. For case 2, the staff assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 60 µCi/gm DEI-131 specified in the Byron and Braidwood TSs.

The major parameters and assumptions used by the staff are provided in Table 5, and the resulting radiological consequence analyses for the EAB and the LPZ and for the control room are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood stations will still provide reasonable assurance that the radiological consequences of a postulated steam generator tube rupture accident will not exceed the dose acceptance criteria specified in SRP Section 15.1.5 and dose guidelines set forth in 10 CFR 100, and the control room dose acceptance criteria specified in GDC 19.

3.5.1.4 Fuel-Handling Accident

The licensee has reevaluated the radiological consequences of a postulated fuel-handling accident (FHA) at an uprated power level of 3658.3 MWt and provided a radiological consequence analysis. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria. A FHA can be postulated to occur either inside or outside of the containment. If the FHA occurs in the containment, the release of fission products can be terminated by closure of the containment based on the detection of high airborne radioactivity. For the postulated FHA occurring outside the containment, the licensee assumed that fission products are released to the environment within a two hour period through the fuel-handling building exhaust system (FHBES). The FHBES is an ESF system that is designed to operate continuously and to bypass the charcoal adsorbers. Upon receiving a high radiation signal, the effluent from fuel handling building is routed through the charcoal adsorbers.

The staff performed the radiological consequences analyses of a FHA assuming a single fuel assembly dropped onto the irradiated fuel stored in the spent fuel pool. The kinetic energy of the falling fuel assembly was assumed to break open the maximum possible number of fuel rods using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (10 percent of noble gases other than krypton-85,
30 percent krypton-85, and 12 percent iodine) was assumed to occur, with the released gases bubbling up through the fuel pool water. The staff assumed an overall effective fuel pool decontamination factor of 100 for iodine and of 1 for noble gases. The staff also provided a 90 percent iodine removal efficiency for the FHBES filter and assumed 1 percent of flow bypassed the filter.

The major parameters and assumptions used by the staff are provided in Table 6, and the resulting radiological consequence analyses are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee. Therefore, the staff concludes that the Byron and Braidwood design will still provide reasonable assurance that the radiological consequences of a postulated fuel handling accident will be well within the dose criteria specified in SRP Section 15.7.4 and the control room dose acceptance criteria specified in GDC 19.

3.5.1.5 Locked Rotor Accident with a steam generator PORV Failure

The reactor primary coolant pump locked rotor accident is caused by an instantaneous seizure of a reactor coolant pump rotor rapidly reducing the primary coolant flow through the affected reactor coolant loop leading to a reactor trip on a low-flow signal. The licensee analyzed this postulated accident assuming that 2 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel gap (10 percent of the core activity) to the reactor coolant. The licensee assumed the primary-to-secondary steam generator tube leak rate is 0.5 gpm for the faulted steam generator and 0.218 gpm for each of the intact steam generators. A steam generator PORV is assumed to fail open resulting in an uncontrolled blowdown of steam from the steam generators directly to the environment for 20 minutes. In addition, radioactivity is assumed to be released to the environment by way of

primary-to-secondary leakage and steaming from the secondary side to the environment. The staff finds these assumptions to be conservative.

The staff has reviewed the licensee's analysis and performed an independent confirmatory dose calculation. The results of the staff's independent radiological consequence calculation are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 8. The radiological consequences calculated by the staff are consistent with those calculated by the licensee.

The staff concludes that the Byron and Braidwood stations operating at an uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed a small fraction the dose guidelines set forth in 10 CFR 100 (30 rem to the thyroid and 2.5 rem to the whole body) and the control room dose acceptance criteria specified in GDC 19.

3.5.1.6 Rod Ejection Accident

The mechanical failure of a control rod mechanism pressure housing is postulated to result in the ejection of a rod cluster control assembly and drive shaft. Because of the resultant opening in the pressure vessel, primary coolant is released to the containment with concurrent rapid depressurization of the reactor pressure vessel. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The licensee assumed that 15 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel-cladding gap of these elements. In addition, the licensee assumed that 0.375 percent of the fuel rods will experience fuel melting. The licensee performed its calculations to obtain these parameters using the guidelines provided in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs," which is acceptable. The staff has reviewed the licensee's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the licensee meet the relevant dose acceptance criteria.

The licensee assumed that the release of fission products to the environment will occur via either one of two pathways. The first pathway involves a release of primary coolant to the containment, which is then assumed to leak to the environment at the design leak rate of the containment. In the second pathway, fission products would reach the secondary coolant via the steam generators with a maximum total allowable primary-to-secondary leak rate of 1 gallon per minute. To verify the licensee's assessments, the staff performed independent radiological consequence calculations for the same two pathways as described above for the control rod ejection accident. The major parameters and assumptions used by the staff are provided in Table 7, and the resulting radiological consequence analyses are provided in Tables 1 and 2 for Byron and Braidwood stations, respectively. The radiological consequences calculated by the staff are consistent with those calculated by the licensee.

The staff concludes that the Byron and Braidwood stations operating at an uprated power level of 3658.3 MWt will still provide reasonable assurance that the radiological consequences of a

postulated rod ejection accident will not exceed a small fraction the dose guidelines set forth in 10 CFR 100 (30 rem to the thyroid and 2.5 rem to the whole body) and the control room dose acceptance criteria specified in GDC 19.

3.5.2 Atmospheric Relative Concentration Estimates

The licensee used five years of onsite meteorological data collected during calendar years 1994 through 1998 to estimate the atmospheric relative concentration (X/Q) values used in the control room dose assessments described above. These data were not used to calculate X/Q values for the EAB and LPZ. In the amendment request, the licensee stated that it used X/Q values that were previously calculated for the EAB and LPZ and are part of the design basis information for the plants.

The 1994 through 1998 meteorological data were measured at 9.1 and 76.2 meters above grade at the Byron site and at 10.4 and 61.9 meters above grade at the Braidwood site. Joint recovery of the wind speed, wind direction and atmospheric stability data during each of the years at both sites was very high, above 99 percent at both levels for all years except in 1998 at Byron, when the upper level measurement recovery was 98 percent. All recovery rates are well above the recommended minimum of 90 percent cited in Regulatory Guide (RG)1.23, "Onsite Meteorological Programs." The licensee confirmed that the meteorological measurement programs met the recommendations of RG 1.23 from 1994 through 1998, as well as currently. The licensee also stated that a contract specialist assists the licensee in managing the meteorological measurements program using its comprehensive field and office procedures manual. Data are downloaded daily and visually checked for accuracy. Equipment conditions are checked during weekly visits to the measurement tower, and tower instrumentation is calibrated on a quarterly basis. Staff performed a review of the data and found fairly good year to year consistency among the data, and between the Byron and Braidwood sites. Wind speed and direction data between the two measurement heights at each site appeared well correlated, as would be expected for these two northern Illinois sites having little local topography.

With respect to the EAB and LPZ X/Q values, the licensee stated that it had used previously calculated values that are part of the design basis for these plants. Staff did not review these X/Q values since there did not appear to be a need to do so. The EAB and LPZ X/Q values are provided in Tables 10 and 11.

The licensee used the ARCON96 methodology described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake," to calculate X/Q values for control room dose assessment. At each site, calculations were made for four postulated release locations for each of the units. Each unit has two control room air intakes, the fresh air intake used during normal operation and the emergency air intake for use in an emergency. In several of the postulated accident scenarios, it was assumed that a short time interval would elapse before outside air intake to the control room would be switched from the fresh air intake to the emergency intake. All postulated releases were calculated as ground level point releases and assumed no effluent flow. One calculation was made for a postulated release location less than 10 meters from the control room fresh air intake. At this time, staff does not recommend use of the ARCON96 methodology at such short distances. However, the calculation was made assuming a point release from the nearest point of the containment to the control room fresh air intake for a period of approximately 2 minutes. This estimate results in a

higher X/Q value than would be calculated at a distance of 10 meters assuming a diffuse release from the containment building. The staff finds the control room X/Q values acceptable. These values and the postulated release location/receptor pairing are provided in Tables 12 through 14.

4.0 SYSTEMS, STRUCTURES, AND COMPONENTS EVALUATION

4.1 Reactor Vessel Integrity

To determine the acceptability of the power uprate on the integrity of the reactor pressure vessel (RPV), the staff evaluated the following:

- effect on the end of life upper-shelf energy (EOL USE) values for beltline materials in the Byron and Braidwood units;
- effect on the licensee's revised pressure and temperature (P-T) limit curves and the licensee's assessment for prevention against pressurized thermal shock (PTS); and
- effect on the material surveillance programs.

4.1.1 Effect on the EOL USE Values for the Byron and Braidwood Units RPV Beltline Materials

Appendix G, "Fracture Toughness Requirements," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 10, Appendix G), requires, in part, that the Charpy-V USEs for RPV beltline materials be no less than 75 ft-lb (102 J) in the unirradiated condition, and no less than 50 ft-lb (68 J) throughout the life of the RPV, unless it can be demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of USE (as determined from the results of Charpy-V tests and Charpy-V curves) will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The staff has reviewed the licensee's projected EOL USE values tabulated in Table 5.1.2-8 of the July 5, 2000, submittal for the beltline materials for Byron Units and Table 5.1.3-8 for Braidwood Units. The staff performed independent EOL USE calculations for the Byron and Braidwood beltline materials. However, upon comparison with the unirradiated USE values currently available in the NRC's Reactor Vessel Integrity Database (RVID), the staff determined that there was some variability in unirradiated USE values reported by the licensee and those currently stored in the RVID for two of the RPV beltline forgings and five of the RPV beltline weld materials. Table 4.1.1-1 below lists these differences.[7]

7 The updated unirradiated USE values are provided in the ComEd letters of dated April 7, 1975, for nozzle forging material 5P-7016 (Braidwood 1) and May 22, 1975, for nozzle forging 5P-7056 (Braidwood 2). The unirradiated USE values for these forgings supersede the values in materials analysis reports dated March 17, 1975, for forging 5P-7016 and May 5, 1975, for forging 5P-70560, which form the current regulatory basis for the unirradiated USE values reported in the Reactor Vessel Integrity Database (RVID). For the beltline weld material heats, the staff used the average values from the values reported by ComEd, and those currently given in the NRC's RVID. The staff will update the RVID to conform to the updated unirradiated USE values the forgings reported in the ComEd letters of April 7, 1975, and May 22, 1975, and which are reported here in Table 2.5.1-1.

Table 4.1.1-1. USE Energy Values for the Byron and Braidwood RPV Beltline Materials

Unit	Weld ID and Heat	RVID	RVID Source	Licensee	Licensee Source	Licensee	Staff	
		Init. USE		Init. USE		EOL USE	Init. USE[1]	EOL USE
Byron 1	WF336 (442002)	74	Response to Request for Additional Information Regarding GL 92-01 Dated November 19, 1993	77	<u>For Forgings 5P7016 and 5P7056:</u> Response to RAI regarding power uprate dated December 21, 2000 <u>For Welds 442002 and 442011:</u> Response to RAI regarding GL 92-01 dated November 19, 1993, and the response to RAI regarding power uprate dated January 31, 2001	69	75 [1]	67
Byron 2	WF447 (442002)	67		80		78	75 [1]	67
	WF562 (442011)	70		80		69	75 [1]	66
Braid-wood 1	Nozzle Shell Forging 5P7016	162		155		132	155 [1]	132
	WF562 (442011)	70		80		75	75 [1]	63
Braid-wood 2	Nozzle Shell Forging 5P7056	128		115		98	115 [1]	98
	WF562 (442011)	70		80		67	75 [1]	63

In its responses to requests for additional information (RAIs), the licensee informed the staff that the unirradiated USE for the five RPV welds in question were obtained in accordance with methods for establishing USE values in ASTM Standard Procedure E185-82. This is an acceptable method because the methods of ASTM Standard Procedure E185-82 are invoked by reference in 10 CFR Part 50, Appendix G. For its independent EOL USE calculations for the beltline welds, the staff applied initial USE values that were based on the arithmetic mean of all initial USE values reported by the licensee and by the staff in the RVID for a given heat of material. The initial USE values used by the staff in its independent USE calculations are provided in the shaded portions of Table 4.1.1-1.

Both the staff's and the licensee's calculations of the EOL USE values are based on the neutron fluence values for the RPV 1/4T locations as determined from the latest neutron transport calculations for the vessels. Since the licensee's fluence values are based on calculated values instead of best-estimated values, the staff concludes that the fluence values are acceptable. However, for some of these beltline materials, the EOL USE values calculated by the staff differed from the EOL USE values calculated by the licensee. The staff determined

that the variation in the EOL USE values resulted from one of two factors: (1) use of different initial USE values in the USE calculations, or (2) a difference in the manner in which the USE surveillance data were applied to the USE calculations. In this case, both the staff and the licensee have confirmed the EOL USE values for the Byron and Braidwood RPV beltline materials will remain above 50 Ft-lb throughout the licensed life of the plants, therefore, the RPV beltline materials for the Byron and Braidwood units will continue to satisfy the EOL USE criteria specified in 10 CFR Part 50, Appendix G, with the 5-percent increase in rated core thermal power.

4.1.2 Effect on the Pressurized Thermal Shock (PTS) Assessments for the Byron and Braidwood RPVs

Section 50.61 to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.61), requires, in part, that "[f]or each pressurized water reactor for which an operating license has been issued, . . . the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. . . For each pressurized water nuclear reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion . . ." [8]

The staff reviewed the following PTS evaluation reports for the Byron and Braidwood Units:

- WCAP-15390 for Byron Unit 1
- WCAP-15389 for Byron Unit 2
- WCAP-15365 for Braidwood Unit 1
- WCAP-15381 for Braidwood Unit 2

In these WCAPs, Westinghouse, acting on behalf of the licensee, demonstrated that the RPVs for the Byron and Braidwood units would continue to satisfy the adjusted reference temperature criteria for pressurized thermal shock (i.e., EOL criteria for RT_{PTS} values) stated in 10 CFR 50.61. As part of its review, the staff performed independent calculations of the projected EOL RT_{PTS} values for the beltline materials in the Byron and Braidwood RPVs based on the projected EOL neutron fluences for the uprated power conditions. For its assessment, the staff used the methodology in 10 CFR 50.61 to calculate projected RT_{PTS} values for these units. Although there were some minor variations in the manner in which some of the chemistry and surveillance data were applied to the EOL RT_{PTS} calculations, both the staff and the licensee confirmed that the beltline materials for the Byron and Braidwood RPVs would still meet the regulatory criteria of the revised PTS rule, 10 CFR 50.61, even under the uprated power conditions for the units. Therefore, the staff concludes both the Byron and Braidwood units will remain in compliance with the criteria of the revised PTS rule, 10 CFR 50.61 even under the uprated power conditions for the units.

8 According to the revised rule, 10 CFR 50.61, the PTS screening criteria are 270 °F for plate materials, forging materials, and axial weld materials, and 300 °F for circumferential weld materials.

4.1.2.1 Effect of the Uprate on the Pressure-Temperature (P-T) Limit Curves for Byron and Braidwood Reactor Coolant Systems

The staff also assessed the licensee's requests for approval of the uprated P-T limit curves for the RCSs, and of the licensee's proposed pressure-temperature limits reports (PTLRs) for the Byron and Braidwood facilities. Holders of licenses for operation of nuclear power generation facilities are required by Section IV.A.2. of Appendix G to 10 CFR Part 50 to establish and implement these P-T limit curves at their respective nuclear plants. Criterion 2 of Paragraph (c)(2)(ii) of Section 50.36 to 10 CFR Part 50 requires licensees to establish a limiting condition for operation (LCO) in their plant-specific technical specifications (TS) for operating restrictions needed to preclude unanalyzed accidents and transients. These operating restrictions include P-T limits and low-temperature overpressure protection (LTOP) limits. Licensees typically incorporate these P-T limit curves and the LTOP system limits into the LCO for the reactor coolant system, and use them as one of the bases for protecting the RPV and reactor coolant pressure boundary (RCPB) against fracture during normal plant operations (including operations during heatups and cooldowns of the reactor and during anticipated operational occurrences), and during pressure testing conditions.

By amendment Nos. 98 and 89 (January 23, 1998), for Byron and Braidwood respectively, the NRC approved license amendment requests that allowed the licensee to remove the P-T limits for the units from the limit conditions for operation in the Byron and Braidwood TS, and incorporate them into a pressure-temperature limits report (PTLR) that would be controlled under a licensee-implemented program that is described in the administrative control section of the TSs. This license amendment was consistent with the staff's administrative guidelines specified in generic letter (GL) 96-03, which provided the staff's position on removing the P-T limit curves from the limiting conditions for operation in the TSs. According to the staff's position stated in GL 96-03, in order to receive NRC approval to relocate the P-T limits to a PTLR , P-T limits must be generated in accordance with the following criteria:

- must comply with the specific requirements of Appendices G and H to 10 CFR Part 50;
- be documented in an NRC-approved topical report or in a plant-specific submittal; and
- be incorporated by reference into the TS (usually by reference in the Administrative Controls Section of the TS)

According to the GL, updates of the P-T limits and LTOP limits that are implemented in accordance with the approved methodology will not need to be submitted for staff review. However, any subsequent changes in the approved methodology will require staff review and approval pursuant to the 10 CFR 50.90 license amendment process.

In the licensee's safety assessment for the power uprate included in the July 5, 2000, license amendment request and supplemented by information given in the licensee submittal of February 20, 2001, the licensee indicated the P-T limit curves for the Byron and Braidwood

RPVs would continue to be generated in accordance with current approved methodology[9] and that any changes to the curves would be implemented through the licensee 10 CFR 50.59 design change process. This is consistent with the staff's position in GL 96-03. Therefore, the staff concludes that the power uprate will not affect Exelon Generation Company's compliance with the criteria of Appendix G to 10 CFR Part 50 or conformance with the staff's position stated in GL 96-03.

4.1.3 Effect on the Material Surveillance Programs for the Byron Units 1 and 2, and Braidwood Units 1 and 2

Appendix H to 10 CFR Part 50 provides the NRC's requirements regarding licensee implemented RPV material surveillance programs. The licensee has provided the surveillance capsule withdrawal schedules for the Byron and Braidwood units in Tables 5.1.2-1, 5.1.2-2, 5.1.3-1, and 5.1.3-2 of the July 5, 2000 submittal. The proposed schedules revised the removal time and the corresponding fluence for the Z capsule from specified values to a "standby" status for each unit. The proposed changes are based on the uprated neutron fluence values for the beltline materials and surveillance capsules. Although the year of the ASTM standard, on which the capsule withdrawal schedule was based, was not mentioned in the submittal, previous surveillance capsule reports indicated that the licensee is using the criteria of ASTM Standard Practice E185-82 as the current basis for the material surveillance programs for the Byron and Braidwood units.[10] The staff verified that the previous three capsules for all units were withdrawn in accordance with E185-82, and the fourth capsule may be classified as "EOL" according to the standard. Designating it as "standby" simply gives additional flexibility to the surveillance programs. Hence, the material surveillance programs with the proposed surveillance capsule withdrawal schedules for the Byron and Braidwood units are consistent with the criteria of 10 CFR Part 50, Appendix H, and are acceptable.

4.1.4 Reactor Coolant Loop Piping and Supports

In Section 5.5 of the July 5, 2000, submittal, the licensee assessed whether the revised conditions resulting from a 5-percent power uprate would adversely effect the LBB status of the reactor coolant loop primary piping at Byron and Braidwood Units 1 and 2. The licensee stated that the input parameters important to the LBB evaluation had been considered and that LBB evaluation had been performed which demonstrated, "that the previous LBB analysis conclusion remains valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the structural design basis of the Byron and Braidwood Units 1 and 2." Based on the changes in operating parameters expected to result from the 5 percent power uprate, the NRC staff concurs with the

9 The current approved methodology approved by the staff for allowing the P-T limits to be controlled under a PTLR and changed under the 10 CFR 50.59 process uses the following bases for generating the P-T curves: (1) the 1989 edition of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code, and (2) the methods of analysis in Westinghouse Topical Report WCAP-14040-NP-A, as modified by (3) the methods of analysis in ASME Code Case N-514. Consistent with the staff position stated in GL 96-03, any changes to these bases as the approved methodology for generating the P-T limit curves will require the licensee to submit a license amendment to change the approved methodology (i.e., submit a license amendment pursuant to the requirements of 10 CFR 50.90).

10 Henceforth ASTM Standard Practice E185-82 will be abbreviated as E185-82.

licensee's conclusion that no change to the LBB status of the Byron and Braidwood Units 1 and 2 reactor coolant primary loop piping is required as a result of the requested 5 percent power uprate.

4.2 Reactor Vessel

The licensee reported that the 5% power increase will result in changing the design parameters given in Tables 2.1-1 and 2.1-2, in Attachment E of the July 5, 2000, submittal. These tables provide a comparison of the current design parameters and the corresponding revised parameters for use in the power uprate analysis at Braidwood and Byron Units 1 and 2.

The licensee evaluated the reactor vessel for the effects of the revised design conditions on the most limiting vessel locations with regard to ranges of stress intensity and fatigue cumulative usage factors (CUFs) in each of the regions, as identified in the reactor vessel stress reports. The evaluations considered the operating parameters which were identified for the uprated power condition. The regions of the reactor vessel affected by the power uprate include outlet and inlet nozzles, the RPV (main closure head flange, studs, and vessel flange), CRDM housing, vessel shell (vessel wall transition, bottom head to shell juncture), core support guides and the instrumentation tubes. The licensee evaluated the maximum ranges of stresses and CUFs for the critical components at the core power uprated conditions. The evaluation was performed in accordance with the ASME III 1971 Edition with addenda through the Summer 1973 for Braidwood and Byron Stations to assure compliance with the code of record.

The calculated maximum stresses and the maximum CUFs for the reactor vessel critical locations are provided in Tables 5.1.1-1 and 5.1.1-2 of the July 5, 2000, submittal for Byron and Braidwood stations respectively. The licensee indicated that all maximum primary plus secondary stress intensities are within the allowable limit of 3Sm except for the RPV inlet and outlet nozzles and the bottom head instrumentation tubes, which were evaluated and justified by simplified elastic-plastic analysis in accordance with NB-3228 in Section III of the ASME Code. The simplified elastic-plastic analysis method is often used in the nuclear industry and acceptable to the staff. The calculated CUFs shown in the tables remain below the allowable ASME Code limit of 1.0.

The licensee concluded that the current design of the reactor vessel continues to be in compliance with licensing basis codes and standards for the power uprate condition. Based on its review, the staff's review concurs with this conclusion.

4.3 Reactor Core Support Structures and Vessel Internals

By letter dated January 31, 2001, the licensee provided the additional information requested by the staff with regard to the evaluation of the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, core barrel, baffle plate, baffle/barrel region bolts, the lower core support structure and the upper core plate. The licensee indicated that because the reactor internal components were designed prior to the introduction of Subsection G of the ASME B&PV Code, a plant specific stress report was not required. However, the design of the reactor internals was designed according to Westinghouse criteria which are similar to the criteria in Subsection G of the ASME Code. The acceptance criteria are the same as used in the original design of the plant and their original licensing basis as documented in the Braidwood and Byron UFSAR.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Tables 2.1-1 and 2.1-2 of the July 5, 2000, submittal for Unit 1 and Unit 2, respectively, at each station. The evaluation indicated that for the lower core plate, the baffle-barrel region components (core barrel, baffle plates, bolts, and former plates) and the upper core plate, the current analyses of record for Braidwood and Byron remain bounding for the power uprate condition. Table 5.2.3-1 of the July 5, 2000, submittal provides the maximum calculated stress and CUF for the most critical component of the lower core support column. The table shows that the maximum stresses and the CUF are less than the allowable limits. The remaining reactor internal components are less limiting. In addition, the potential for the flow induced vibration does not increase for the power uprate. As a result of these evaluations, the licensee concluded that the reactor internal components at Byron and Braidwood Stations will be structurally adequate for the proposed power uprate conditions. The staff concurs with the licensee's assessment.

4.4 Control Rod Drive Mechanisms (CRDM)

The pressure boundary portion of the CRDMs are those exposed to the vessel/core inlet fluid. The licensee evaluated the adequacy of the CRDMs by reviewing the Byron and Braidwood Stations current CRDM design specifications and stress report to compare the design-basis input parameters against the revised design conditions in Tables 2.1-1 and 2.1-2 of July 5, 2000, submittal for the power uprate. Table 5.4-1 of the submittal shows that the current design basis conditions for the CRDMs are bounding for the power uprate. The licensee's January 31, 2001, submittal identifies the applicable ASME Code and results of the stress and fatigue evaluation for the CRDM components. The licensee indicated that the Code used for the power uprate evaluation is the ASME B&PV Code Section III, 1974 Edition through Summer 1974 Addenda, which is the Code of record. The analytical results provided by the licensee indicate that CRDM components' stresses and CUFs for the proposed conditions remain within the ASME Code limits.

On the basis of its review, the staff concurs with the licensee's conclusion that the current design of CRDMs continues to be in compliance with licensing basis codes and standards for the power uprated conditions.

4.4.1 CRDM Nozzles

In its submittal of December 11, 1998, the Nuclear Energy Institute (NEI) provided the NRC with a relative probabilistic susceptibility ranking of CRDM nozzles in domestic PWRs to initiate and grow flaws induced by primary water stress-corrosion cracking (PWSCC). In this submittal, NEI and the Materials Reliability Project (MRP) projected that the CRDM nozzles at Farley Unit 2, North Anna Unit 1, Surry Unit 1, DC Cook Unit 2, Point Beach Unit 1, Ginna, and Diablo Canyon Unit 2 would be among the CRDM nozzles that are more highly susceptible to PWSCC, and projected that the CRDM nozzles for the Byron and Braidwood RPVs would be significantly less susceptible to PWSCC than those in the aforementioned plant designs. The NEI/MRP integrated program for managing postulated PWSCC in the CRDM nozzles of domestic PWRs calls for voluntary volumetric examinations to be conducted at the nuclear facilities that are considered to have some of the more highly ranked CRDM nozzles in the PWR-industry. For Westinghouse designed PWRs, voluntary volumetric examinations have been completed on the CRDM nozzles of the DC Cook, Unit 2, North Anna, Unit 1, and Ginna nuclear Power plants. The Southern Nuclear Operating Company and the Pacific Gas and Electric Company

have also committed to inspect the CRDM nozzles of the Farley, Unit 2 and Diablo Canyon, Unit 2 nuclear plants as part of the NEI/MRP integrated for managing postulated PWSCC in PWR vessel head penetrations (VHPs); these inspections are currently scheduled to occur in 2002 and 2004. NEI and the MRP have indicated that they will use the results of the voluntary volumetric examination initiatives, as well as the data from any reported CRDM nozzle leakage events, as the basis for both evaluating the need to revise the susceptibility modeling and rankings, and the need to conduct additional voluntary, volumetric inspections of the CRDM nozzles at other facilities.

The bases for increasing the power of the Byron and Braidwood units are consistent with the those approved by the NRC as the basis for increasing the power for the Farley units in 1998. The licensee has not committed to conducting any volumetric examinations of the CRDM nozzles of the Byron and Braidwood nuclear plants at this time. However, because the bases for increasing the power of the Byron and Braidwood units is consistent with those previously reviewed and approved for the power increase for the Farley units, the staff considers that the NEI/MRP integrated program will continue to be a sufficient basis for evaluating the susceptibility of the CRDM penetration nozzles at the Byron and Braidwood nuclear plants to develop PWSCC. The staff will use the results of the Farley CRDM penetration nozzle examinations, as well as any generic CRDM penetration nozzle leakage history,[11] as the basis for evaluating the CRDM penetration nozzles of the Byron and Braidwood units in the future.

4.5 Steam Generators

The licensee has replaced the original steam generators in the Byron Unit 1 and Braidwood Unit 1 plant designs. The heat exchanger tubes in the replacement steam generators are made of alloy 690 and the tube sheet is made of stainless steel. No significant degradation has been observed so far. The licensee evaluated the effects of the proposed power uprate for the BWI RSGs in Section 5.7.1 of the July 5, 2000, submittal.

Byron Unit 2 and Braidwood Unit 2 are currently designed with the original Model D5 steam generators that were installed during initial plant fabrication. The heat exchanger tubes in the Model D5 steam generators are made of thermally treated Alloy 600, and have a nominal outside diameter of 0.75 inch and a 0.043 inch nominal wall thickness. The Model D5 steam generator tubesheet is designed with full-depth, hardrolled expansion joints. The support plates are made of 405 Stainless Steel with a drilled hole configuration. D5 power SG uprate evaluations are addressed in Section 5.7.2 of the submittal.

The licensee reviewed the existing structural and fatigue analyses of the SGs at Byron and Braidwood Stations, and compared the power uprate conditions with the design parameters of both the BWI RSG and the Model D5 SGs stress reports. The comparison of key parameters for the original power uprate conditions is shown in Tables 2.1-1 and 2.1-2 of the July 5, 2000, submittal. The analysis input parameters for the BWI RSG structural evaluation is given in Table 5.7.1.1-1, which contains the same values as those in Table 2.1-1 for the uprated power

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11 The NEI/MRP integrated inspection program CRDM penetration nozzles calls for both the inspection results of the voluntary volumetric inspection initiatives and the results of any reported CRDM penetration nozzle leakage events to be evaluated with respect to their effect on the susceptibility modeling bases and rankings for the industry.

level. As such, the Byron and Braidwood Unit I RSGs were analyzed at the uprated power conditions. The evaluation for BWI RSG was performed to the requirements of the ASME Boiler and Pressure Vessel Code Section III, 1986 edition with no addenda which is the Code of record for BWI SGs at Braidwood Unit 1 and Byron Unit 1.

For evaluation of the critical components of Model D5 SGs , the licensee incorporated the key input parameters in the development of the scaling factors shown in Table 5.7.2.1-1 and 5.7.2.1-2 for the primary and secondary sides, respectively, over the applicable transients. For primary side components, the scaling factors are ratios of primary to secondary pressure differentials for current operating and uprated conditions. For secondary side components, the scaling factors are ratios of secondary pressures for current operating and power uprate conditions. The scaling factors were used to calculate the stress and fatigue usage for the power uprate conditions. The evaluation for the Model D5 power uprate was performed in accordance with the requirements of the ASME Code, Section III, 1971 Edition through the Summer 1972 Addendum, which is the Code of record for Model D5 SGs at Braidwood Unit 2 and Byron Unit 2. The staff finds the licensee's evaluation methodology to be conservative and, therefore, acceptable.

The calculated maximum stresses and CUFs for the critical SG components are provided in Tables 5.7.1.1-3 to 5.7.1.1-5 of the July 5, 2000, submittal for BWI RSGs and in Tables 5.7.2.1-3 to 5.7.2.1-6 for Model D5 SGs. The results indicate that the maximum calculated stresses are below the Code-allowable limits, and the calculated CUFs are within the allowable limit of unity for the 40 years service life. In addition, the licensee performed flow-induced vibration (FIV) analysis for U-bend tubes to determine the tube vibration response following the power uprate. As a result, the licensee concluded that the fluid velocities were found to be less than the critical velocity and that there was no increase in the potential for the FIV during the power uprate. The staff agrees with this conclusion.

On the basis of its review, the staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the critical SG components to be within the Code allowable limits and, therefore, acceptable for the proposed power uprate at Braidwood and Byron stations.

4.5.1 SG Tube Integrity

4.5.1.1 Evaluation of Unit 1 Steam Generator Tube Degradation Mechanisms

In its November 27, 2000, response to the staff's RAI No. D.2 regarding the effect of the power uprate on antivibration bar (AVB) wear, the licensee provided a summary of its operational assessments for AVB wear that demonstrated that the existing allowable operating interval between inspections will remain the same. These operational assessments further demonstrated that performance criteria are satisfied for the inspection interval, after considering uprate conditions. These assessments will be updated to reflect any planned inspections performed prior to implementing the power uprate.

With regard to the operating parameters affected by the power uprate, corrosion of steam generator tubing is sensitive to T_{hot}. For both units, the licensee indicated that T_{hot} will be increased from 610 to 617 °F after core thermal power uprate is implemented; the primary to secondary pressure differential will be decreased from 1252 to 1215 psi, a net drop of 37 psi.

The licensee also evaluated the effect of power uprate to tube degradation and stated that the uprate will have a negligible impact on tube degradation. Industrial experience with Alloy 690 tubing at these temperatures has been good. On the basis of this experience and the licensee's steam generator program for ensuring tube integrity between steam generator inspections, the staff concludes that the power uprate will not adversely impact tube integrity for the Byron and Braidwood Units 1 SGs.

4.5.1.2 Evaluation of Unit 2 Steam Generator Tube Degradation Mechanisms

In its response to the staff's RAI No. D.2 regarding the effect of the power uprate on AVB wear, the licensee provided a summary of its operational assessments for AVB wear that demonstrated that the existing allowable operating interval between inspections will remain the same. These operational assessments further demonstrated that performance criteria are satisfied for the inspection interval, after considering uprate conditions. These assessments will be updated to reflect any planned inspections performed prior to implementing the power uprate.

With regard to the operating parameters affected by the power uprate, corrosion of steam generator tubing is sensitive to T_{hot}. For Byron Unit 2 and Braidwood Unit 2, T_{hot}, will stay the same at 611°F after core thermal power uprate is implemented. The secondary side pressure will decrease to make the primary to secondary pressure differential increase from 1327 to 1340 psi, a net change of 13 psi. The licensee's analysis shows that power uprate will have almost little effect on tube degradation based on the key factor that T_{hot} stays the same. As discussed above, the staff also evaluated the effect of the operating parameter changes associated with the power uprate and concludes that the power uprate will not significantly impact tube degradation for the Byron and Braidwood Units 2 SGs.

4.5.1.3 Steam Generator Tube Inspection

With respect to SG tube inspection, the licensee stated that the 5-percent power uprate has not affected the degradation assessment, therefore, the licensee will not change the inspection plan for the upcoming outage. The licensee stated that future inspections will be determined by active degradation, potential degradation, industry experience, and plant-specific operating experience. On the basis of this experience and the licensee's steam generator program for ensuring tube integrity between steam generator inspections, the staff concludes that the power uprate will not adversely impact tube integrity.

4.5.2 Steam Generator Tube Plugging and Repair Criteria

The current plugging limit for tube degradation in the Byron and Braidwood TSs is 40 percent of the wall thickness. In general, tubes are plugged on detection. Any detected tube indication for degradation by thinning or wear that is less than 40 percent throughwall is allowed to remain in service in accordance with the TSs. Both of these degradation types can be bound by uniform wall-thinning calculations. The licensee performed wall-thinning calculations for degraded tubing in accordance with RG 1.121, which specifies that the tube should maintain a safety margin of three with the primary-to-secondary pressure differential under normal operating conditions. The licensee's calculations showed that the plugging limit of 40 percent for tube degradation is conservative under the pressure loading of $3\Delta P$ in the power uprate condition.

The current plugging limit for laser welded sleeves in the Byron Braidwood TS is 40 percent of the wall thickness. This limit will be reduced to 38.7 percent after the power uprate. The plugging limit for TIG welded sleeves will remain at 32 percent. Sleeves with crack-like indications would be plugged since there are no qualified sizing techniques. Any sleeve indications of degradation by thinning or wear that are less than these limits are allowed to remain in service in accordance with the TS. Both of these degradation types can be bound by uniform the wall-thinning calculations. The licensee performed sleeve wall-thinning calculations for degraded sleeving in accordance with RG 1.121, which specifies that the sleeve should maintain a safety margin of three under the primary-to-secondary pressure differential under normal operating conditions. The licensee's calculations showed that the plugging limit of 38.7 percent for laser welded sleeve and 32 percent for TIG welded sleeve degradation are conservative under the pressure loading of 3ΔP in the power uprate condition.

In a response to the staff's RAI D.1, regarding the effect of the uprate on the steam generator plugging criteria, the licensee stated that power uprate will not result in any changes to the eddy current measurement errors, and changes to continuing degradation growth rates have been identified for evaluation of the TS repair limit. The licensee has stated that the sufficient repair limit allowances exist to account for the eddy current measurement uncertainty and any projected degradation growth under uprated power conditions. These allowances will ensure that the tubing structural limit is not exceeded.

The staff concludes that the existing 40-percent plugging limit for tube degradation, the plugging limit of 38.7 percent for laser welded sleeves, and the plugging limit of 32 percent for TIG welded sleeves in the Byron/ Braidwood TS are adequate under the uprated power conditions

4.5.3 Steam Generator Blowdown (SGBD) System

The steam generator blowdown (SGBD) system is used to control chemical composition and buildup of solids in the steam generator shell water. The SGBD systems in the Byron and Braidwood plants are designed to handle a maximum continuous blowdown rate of 90 gpm for each steam generator. The actual blowdown flow during plant operation depends on the type of chemistry control and the requirements for controlling solid buildup on steam generator tubesheets. The SGBD system is designed for the highest pressure setpoint in the main steam safety valve which does not change with power uprate. Also, its operating temperatures at power uprate are bounded by the system design. Consequently, the range of normal blowdown flow after power uprate will remain within the recommended range of 0.2 percent to 1 percent of steam generator steam flow. Based on its evaluation the licensee concluded that the proposed power uprate will not adversely affect performance of the SGBD in the Byron and Braidwood plants. The staff reviewed the licensee evaluation and finds it acceptable.

4.6 Reactor Coolant Pumps (RCPs)

The licensee evaluated the existing design basis analyses of the Byron and Braidwood Stations Westinghouse Model 93A RCPs against the revised design conditions for the power uprate as shown in Tables 2.1-1 and 2.1-2 of the July 5, 2000, submittal. The licensee indicated that the evaluation was performed in compliance with the original design specifications and the ASME Code, 1971 Edition with addenda through Winter 1972, which is the Code of record.

At Braidwood and Byron Stations, after the proposed power uprate, the reactor coolant system (RCS) pressure remains unchanged. The most limiting RCP design parameter of the SG outlet temperature decreases slightly from 558.4 to 555.7 °F. There are no changes to the design transients and number of cycles shown in Table 5.6.-2 of the submittal for all service conditions. Table 5.6-3 summarizes the calculated maximum stresses and CUFs for the critical RCP components including pump casing, main flange, thermal barrier flange and main flange bolts. The results indicated that the maximum stresses and CUFs for the power uprate condition for the Byron and Braidwood RCPs are less than the code allowable limits.

On the basis of its review, the staff concurs with the licensee's conclusion that the current RCPs, when operating at the proposed uprated power conditions will remain in compliance with the requirements of the codes under which the Byron and Braidwood Stations were originally licensed

4.7 Pressurizer

The licensee evaluated the structural adequacy of the pressurizer and components for limiting locations at the pressurizer spray nozzle, the surge nozzle, and the upper shell for operation at the uprated conditions. The Code used in the evaluation is the ASME Code, Section III, 1971 Edition, through Summer 1973 addenda, which is the Code of record for Braidwood and Byron Units 1 and 2 pressurizers. The evaluation was performed by comparing the key parameters in the current Byron and Braidwood pressurizer stress report against the revised design conditions in a table on page 5-128 of the July 5, 2000, submittal. The table provides the comparison of pressurizer design parameters for the current operation, uprated power, and design basis condition. The comparison shows that the design basis analyses remain bounding for the proposed power uprate conditions. The licensee concluded that with RCS pressure remaining unchanged the existing pressurizer components will remain adequate for plant operation with the proposed power increase. The staff agrees with the licensee's conclusion.

4.8 Chemical and Volume Control System (CVCS)

The main role of the chemical and volume control system (CVCS) is to manage RCS water inventory, boron concentration and water chemistry. In order to perform these functions, the CVCS must meet the following requirements: (1) the portions of the system that constitute the reactor coolant pressure boundary (RCPB) must be capable of withstanding the expected RCS conditions, (2) introduction of boron into the RCS must meet the design requirements for reactivity control, and (3) with the exception of reactor coolant pump seal injection line, the system must be capable of automatically isolating during all events requiring containment isolation. The proposed power uprate will not affect the CVCS isolation function, but it may have some effect on the integrity of the RCPB and on the boration of the RCS.

4.8.1 RCS Temperature

The licensee has performed an analysis of the CVCS performance after power uprate. The results of the analysis indicated that the temperature of the incoming coolant from the RCS cold leg is between 541.7 °F and 555.4 °F, which is below the current temperature of 558.1 °F, and well below the design and operating temperatures for the regenerative and excess letdown heat exchangers (640 °F and 560 °F, respectively). Similarly, the inlet temperature of water in the

letdown heat exchanger is bounded by the existing design temperature of 400 °F and operating temperature of 288.7 °F. The outlet temperature of the letdown heat exchanger is controlled by an instrument which adjusts component cooling water flow and maintains temperature at a preset level. Since the uprated CVCS temperatures are either bounded by the existing temperatures or controlled at preset levels by the plant operators, the licensee concluded that the power uprate will have no adverse effect on the design and operation of the CVCS. The staff reviewed the licensee's evaluation and concludes that it is acceptable.

4.8.2 Boration

See Section 3.2.4 of this SE.

4.8.3 Boron Recycle System

The boron recycle system (BRS) is a plant system that is aligned to the CVCS and is designed to accept and process all effluents that can be readily recycled to the RCS. BRS receives letdown flow from the CVCS downstream of the letdown heat exchanger for processing. Since the RCS cold leg temperature under uprated power conditions will be lower than the temperature specified in the current design basis, the temperature of the letdown water will not exceed its preset value. The licensee concluded, therefore, that the operation of the BRS will not be affected by the power uprate. The staff concurs with the licensee's conclusion.

4.9 NSSS Piping and Pipe Supports

The licensee evaluated the NSSS piping and pipe supports by reviewing the design basis analysis against the uprated power condition, with regard to the design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, RCL branch piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The methods, criteria and requirements used in the existing piping analysis design specification described in the Byron and Braidwood UFSAR were used for the power uprate evaluation. The evaluations are based on the requirements of the ASME B&PV Code, Section III, 1974 Edition up to and including Summer 1975 Addenda, and other later ASME Code Edition listed in the UFSAR, such as 1977 Edition up to and including Summer 1979 Addenda, which was used for stress analysis of reactor loop piping and branch nozzles. However, the ASME B&PV Code, Section III, 1986 Edition was used for the stress analysis of the pressurizer surge line due to the stratification loading condition.

The RCS pressure will remain unchanged for the proposed core power uprate. The actual hot leg temperature for the power uprate is projected to be slightly greater than the hot leg temperature at the current rated power level. The cold leg temperature for the power uprate conditions will be less than that for the current power level. The licensee indicated that there is sufficient margin in the existing analysis for stresses associated with the temperature changes defined in Tables 2.1-1 and 2.1-2 of the July 5, 2000, submittal, for Unit 1 and Unit 2, respectively, at each station.

The licensee also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for Byron and Braidwood Stations power uprate. The loop hydraulic forces will increase slightly due to the decrease in the cold leg

temperature and the increase in water density at the power uprate condition. The licensee indicated that the small increase in LOCA loads for the power uprate is offset by the application of LBB which excludes the dynamic effects associated with the original design basis postulated pipe ruptures of the primary loop piping. With the application of LBB, the LOCA loads of the current licensing basis were reduced based on the less severe branch line breaks, such as in the accumulator line, pressurizer surge line, and residual heat removal line. As such, the design basis LOCA hydraulic forcing functions are bounding for the uprated power condition. The licensee concluded that the existing stresses, fatigue usage factors and loads will continue to meet the ASME Code requirements for the power uprate. The staff agrees with the licensee's conclusion.

In its submittal of January of 31, 2001, the licensee provided, in Tables J.5-1 through J.5-6 the calculated maximum stresses, fatigue usage factors and loads. The values of the maximum stresses, CUFs, and loads are less than the corresponding allowable limits for the power uprate for the NSSS components including the reactor cooling loop piping, the RCS branch nozzles, the reactor pressure vessel supports, and the primary equipment (including reactor coolant pump, steam generators, and the pressurizer) supports. The licensee reviewed the design basis parameters affected by the power uprate (i.e., the hot leg temperature is unchanged at 618.4 °F, and the cold leg temperature decreases from 558.4 °F to 555.7 °F for the power uprate), and found the original piping analysis loads to be bounding for the power uprate for the primary equipment nozzles, and the pipe supports. The staff concurs with the licensee's conclusion that these components will continue to be in compliance with the Code of record at Braidwood and Byron Units 1 and 2, and are therefore, acceptable for the power uprate.

4.10 NSSS/BOP Interface Systems

4.10.1 Auxiliary Feedwater System and Condensate Storage Tank

The licensee performed evaluations of the effects of plant operations at the proposed uprated power level on the auxiliary feedwater (AFW) system and condensate storage tank (CST). It was determined that the AFW system flows for various transients and accidents are acceptable for plant operations at the proposed power level.

The condensate inventory required for the limiting transient and accident conditions was determined to be 198,619 gallons. Currently, both Byron and Braidwood maintain their CSTs at a minimum usable volume of 200,000 gallons.

The current Byron and Braidwood licensing basis requires that sufficient CST inventory must be available to bring the unit from full power to hot standby conditions under natural circulation conditions, maintain the unit at hot standby for four hours, and then cool the RCS to the residual heat removal system cut-in conditions within four hours. The results of the licensee's evaluation for the power uprate conditions concluded that the current TS limit of 200,000 gallons in the CST is sufficient to meet the above stated licensing basis requirement. The staff agrees with the licensee's assessment and finds it acceptable.

Based on the staff's review and the experience gained from staff review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level will have a little impact on the AFW system and condensate storage tank.

4.10.2 Component Cooling Water System (CCWS)

The CCWS is a closed loop system which serves as an intermediate barrier between the essential service water system, and the systems and components which contain radioactive or potentially radioactive fluids. It provides cooling water to various safety and non-safety systems during all phases of normal plant operation, including startup through cold shutdown and refueling, as well as following a station black-out event, LOCA or MSLB accidents. The CCWS heat loads resulting from plant operations at the proposed uprated power level will increase slightly. The increased heat loads due to power uprate are primarily due to the increased spent fuel pool heat load, residual heat removal (RHR) system heat load during plant cool down, and RHR heat load during post LOCA recirculation mode. The licensee performed evaluations of the effects of these increases in heat loads on CCWS and concluded that the existing CCWS has the capacity to accommodate the slight increase of heat loads resulting from the power uprate with no equipment changes required.

Based on the staff's review and the experience gained from staff review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the CCWS and have an insignificant or no impact on the CCWS. Therefore, the staff concludes that the CCWS is acceptable for Byron and Braidwood operations at the proposed uprated power level.

4.10.3 Spent Fuel Pool Cooling System (SFPCS)

The SFPCS is designed to remove the decay heat released from the spent fuel assemblies stored in the spent fuel pool (SFP), to maintain the SFP water temperature at or below the design operating temperature limit of 150 °F during plant operations and refueling, and to maintain its cooling function during and after a seismic event. The SFPCS heat loads will increase slightly resulting from plant operations at the proposed power level.

In the response (dated December 21, 2000) to the staff's request for additional information (RAI), the licensee provided the calculated SFP temperature[12] as a function of time during planned refueling outages to reflect the increase of decay heat in the SFP due to plant operations at the proposed power level. The peak calculated SFP temperature increases from the previous[13] calculated temperature of 157.13 °F to 162.7 °F. The SFP design operating temperature limit of 150 °F is exceeded for a duration of approximately 200 hours. The licensee performed evaluations to demonstrate the acceptability of SFPCS operation, SFP liner and concrete structure at SFP temperatures in excess of 150 °F, and concluded that no changes to the SFP cooling systems are required to support plant operations at the proposed power uprate level. The staff's acceptability of SFP operating temperatures in excess of the design operating temperature limit of 150 °F for SFP liner and concrete is addressed in Section 4.10.3.1 of this SE.

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12 For the bounding case - a full-core offload with one SFP cooling train in operation.

13 In the previous SFP thermal hydraulic analyses submitted in March 1999, for SFP re-rack application request, the calculated peak SFP temperature was 157.13 °F. The SFP design operating temperature limit of 150 °F was exceeded for a duration of approximately 120 hours.

During a conference call with the staff on February 14, 2001, the licensee stated that plant operating procedures have provisions to ensure that both trains of SFP cooling system are available and operable prior to core offload during a planned outage.

Based on the review of the licensee's rationale, the staff finds that plant operations at the proposed uprated power level do not change the design aspects and operations of the SFPCS, and Byron and Braidwood will have reliable SFPCS for cooling the SFP during planned refueling outages. Therefore, the staff concludes that the SFPCS is acceptable for operations at the proposed uprated power level.

4.10.3.1 SFPCS Resin Beds

The 5 percent increase in power may increase the rate of depletion for the cleanup resins in the cleanup beds of the spent fuel pool cooling system and therefore may increase the frequency that the resin beds need to be replenished. However, the licensee's control room and spent fuel system operators monitor these resin beds for the pressure drop (ΔP) across the resin beds. Any significant increase in the ΔP level across the resin beds above a preset replacement criterion level is an indication to the licensee's operators that the resin beds need to be replaced. Since the need to replace resins is controlled by plant operators in accordance with operational criteria that are defined in plant operations procedures, the staff concludes that the proposed increase in power will not have any significant effect on the impurity levels in the spent fuel coolant for the Byron and Braidwood units.

4.10.3.2 Spent Fuel Pool Structural Integrity

In Table H-2, of the December 21, 2000, submittal, the licensee summarized the acceptance criteria for SFP temperature given in the SRP, the analysis results given in the current UFSAR, and the proposed power uprate condition. The licensee indicated that the full core offload is considered a temporary condition during refueling since two-thirds of the core will be routinely returned to the reactor vessel approximately four days following core offload. With a single active failure, the calculated bulk SFP water temperature exceeds the SRP limit of140 °F during this temporary condition but remains below 140 °F with two trains of SFP cooling.

A full core off-load will produce a maximum bulk pool temperature of 162.7 °F assuming a single active failure resulting in the loss of one train of SFP cooling. The calculated SFP temperature exceeds the SRP guidance for approximately four days. In addition, the licensee indicated that the concrete temperature will not be uniformly elevated to the maximum bulk pool temperature of 162.7 °F and the average temperature associated with this gradient will be below the ACI limit of 150 °F. The temperature of 162.7 °F was calculated using conservative assumptions and was based on the final fuel off load with the SFP filled to capacity. For the case of a full core discharge with two heat exchangers operable, the maximum temperature 100 hours after shutdown would be 133.8 °F. Therefore, the temperatures during a normal refueling are not expected to peak above 140 °F. The licensee also indicated that the SFP temperature alarm is set at 149 °F to alert operators of abnormal condition, such as a loss of SFP cooling.

The staff finds that the impact of the maximum bulk pool temperature of 162.7 °F for approximately four days will be minimal with negligible effects on the concrete structure considering that the SFP temperature alarm is set at 149 °F at Braidwood and Byron stations,

that provides an additional precaution to alert the operator for the condition with respect to the American Concrete Institute (ACI) limit of 150 °F. In addition, the licensee performed an analysis that confirmed the maximum rebar stress of 53.7 ksi for the maximum bulk pool temperature of 162.7 °F to be within the allowable limit of 54 ksi. Therefore, the staff concludes that the existing spent fuel pool structures is adequate and acceptable for the proposed power uprate condition at the Braidwood and Byron nuclear stations.

4.11 Main Turbine Generator

4.11.1 Main Turbine

The licensee performed evaluations on turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by plant operations at the proposed uprated power level. Results of the evaluations showed that there would be no increase in the probability of turbine overspeed. Therefore, the turbine could continue to be operated safely at the proposed uprated power levels.

4.11.2 Main Turbine Auxiliary Systems

The licensee stated that performance of the turbine auxiliary systems (i.e. moisture separator, gland sealing steam systems, lube oil system, turbine steam piping system, etc.) were evaluated for power uprate. The licensee determined that these systems are adequate for plant operations at the proposed uprated power level.

Since these systems do not perform any safety function and their failure will not affect the performance of any safety-related system or component, the staff did not review the impact of plant operations at the proposed uprated power level on the designs and performances of these systems.

4.12 High Energy Line Break (HELB) Outside Containment

The licensee evaluated the system operating parameters for plant operations at the proposed uprated power level against the system pressure and temperature parameters used in the existing plant bases to demonstrate the acceptability for HELB effects. The licensee stated that the power uprate will not change the bounding temperature and pressure used as the basis for pipe break analyses. The design basis analyses remain bounding for all HELB events.

Based on the staff's review and the experience gained from staff review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level have an insignificant or no impact on the consequences (e.g. environmental pressure and/or temperature parameters, etc.) resulting from HELB outside containment.

4.13 Safety-Related Equipment Qualification (EQ)

The licensee evaluated the effects of all changes due to plant operations at the proposed uprated power level on design and EQ of mechanical components. The temperatures, pressures, and in some cases flows, in certain systems would be affected slightly by plant operations at the proposed uprated power level. However, these changes in temperatures,

pressures and flows are bounded by the original design of components. The licensee determined that existing parameters used for qualifying mechanical components inside and outside containment remain bounding for the conditions resulting from plant operations at the proposed power level.

Based on the staff's review, it finds that plant operation at the proposed uprated power level will have an insignificant or no impact on the EQ of safety-related mechanical components inside or outside the containment and, therefore, is acceptable.

4.13.1 Radiological Doses

In accordance with 10 CFR 50.49, safety-related equipment must be qualified to survive the radiation environment at its specific location during normal operation and during an accident. The staff evaluated the impact of the power uprate on the safety-related electrical equipment EQ at both post-accident conditions as well as during normal operation.

The licensee compared the power uprate total integrated doses to the original environmental dose established for each environmental radiation zone (UFSAR Table 3.11-2). This comparison indicated that the existing values have sufficient margin to envelop the impact of power uprate. For safety-related equipment for which location-specific environmental doses had been calculated, the licensee compared the power uprate doses to the qualification doses used for the individual component or piece of equipment. This comparison showed that there is sufficient margin to accommodate the increase due to the power uprate without compromising equipment qualification. Since radiological cumulative dose was either enveloped by the original environmental dose established for each environmental radiation zone or was within the threshold limit for which the individual components of equipment were qualified, the staff concludes that the power uprate does not impact the radiological EQ in post-accident conditions.

The staff also reviewed the licensee's evaluation on the impact of power uprate during normal operation. Based on Section 9.4 of the July 5, 2000, submittal, the normal operation component of the total integrated doses used for radiological EQ qualification is not impacted by the power uprate.

4.13.2 Containment Pressure and Temperature Elevation

The licensee evaluated the accident temperature profile inside containment at the uprate power conditions. The licensee compared the inside containment temperature profiles to the existing bounding profiles. The licensee determined that existing profiles remain bounding.

The staff requested the licensee to confirm that the uprate accident pressure profile inside containment is enveloped by the existing design-basis pressure profile. In response to the staff's request, the licensee stated that the test pressure used in the Byron and Braidwood EQ programs bounds the containment design pressure of 50 psig, and therefore bounds the calculated peak pressured determined for the design-basis accidents (i.e., LOCA, MSLB) under power uprate conditions. For pressure, qualification acceptability is determined by comparing the pressure tested in the EQ program to the calculated peak pressure. If the tested pressure value exceeds the calculated peak pressure value, the qualification is acceptable. For EQ purposes, pressure effects are not time dependent. If the peak pressure has been addressed,

so have lower pressures. Since the uprate accident pressure profile inside containment is bounded by the existing design-basis pressure profile, the electrical equipment located inside containment remain qualified for the accident pressure environments at the uprate conditions.

In summary, the staff finds that electrical equipment located inside and outside the containment which performs a safety-related function remains qualified for the accident temperature and pressure environments at the uprate power conditions.

4.13.3 Surface temperature analyses

The license stated that the power uprate will result in revised containment pressure and temperature profiles for the LOCA and the MSLB events. The outside containment MSLB event will also result in revised temperature profiles for the main steam piping tunnels and the associated valve enclosures.

The licensee also evaluated the temperature profiles outside containment at the uprate power conditions. This evaluation showed that the peak temperature (413.5 °F) prior to main steam isolation exceeds the current maximum of 373 °F but remains below the temperature of 419 °F previously used to demonstrate equipment qualification. The peak long-term temperature used for post-accident monitoring equipment outside containment in the steam tunnels and valve room exceeded the current peak temperature of 515 °F by 3 °F. However, the licensee stated that the post-accident monitoring equipment remains qualified. The staff requested the licensee to explain how the post-accident equipment remains qualified when the peak long-term temperature exceeds the current peak temperature. The staff also requested the licensee to give the temperature for which a typical post-accident monitoring equipment was previously qualified.

In response to the staff request, in the letter dated January 31, 2001, the licensee stated that operation at 518 °F would have no impact on (EQ) because of the significant margin between the test and postulated plant conditions. For example, the ITT Barton steam generator transmitters have two functions. Their active safety function is to provide the main steam isolation signal or safety injection signal. The steam generator pressure transmitters must operate during the MSLB to transmit the low-pressure signal that closes the main steam isolation valve (MSIV) a few seconds after the pressure test point is reached. The manufacturer qualified the transmitters, via type test, up to a peak temperature of 486 °F. By a significant margin, this temperature bounds the 414 °F power uprate temperature at the time of MSIV closure for power uprate. The second function of the transmitters is to provide post accident monitoring indication in accordance with a Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environmental Conditions During and Following an Accident." The manufacturer conducted a test on the transmitter up to superheated steam impingement conditions. The recorded temperature on the transmitter surface was 635 °F. This temperature envelops the peak long-term post-accident temperature of 518 °F, as determined by power uprate analysis, by a significant margin. The licensee stated that this qualification is typical of the qualifications performed for the Class 1E electrical equipment in the steam tunnel and valve rooms. Since the revised temperature profiles are still bounded by the EQ test curves and because there are significant margins between the test and postulated plant conditions, the staff concludes that the electrical equipment located inside and outside the containment remain qualified for the temperature environments at the uprate power conditions.

4.14 Safety/Relief Valves

At Byron and Braidwood Stations, the analyses were performed at a 103 percent of the relief valve lifting setpoint for the power uprate. The relief valve setpoints, rated capacities and corresponding dynamic loads due to valve operation imposed in the piping and adjacent structures did not change as a result of the power uprate. On this basis, the staff finds the safety and relief valves will continue to perform their function at the power uprate conditions.

The staff has evaluated the adequacy of the pressurizer safety valve sizing at the uprated power level. It finds that the scenario assumed in the loss of load transient analysis of for sizing the pressurizer safety valves is the same as the loss-of-load analyses performed in the non-LOCA transient analysis for Byron and Braidwood. Since the results of the loss of load analysis performed in the non-LOCA analysis demonstrated that all acceptance criteria are met at uprate conditions, the staff concludes that the current pressurizer safety valve size remains adequate at the uprated power level

4.15 Reactor Trip System/Engineering Safety Feature Actuation System Instrumentation Trip Setpoints and Allowable Values

In its submittal, the licensee stated that its analyses for the power uprate found actuation setpoints and allowable values of the RTS and the ESFAS functions to be within acceptable limits, and therefore, the actuation setpoints and allowable values of these safety systems need not be revised for the uprated power operation. The results of the NSSS analysis demonstrated that all safety-related systems were capable of performing their current design-basis functions, either without changes or with appropriate minimal changes to few alarms to compensate for the effects of the power uprate on the monitored process variables.

In its submittal, the licensee stated that the RTS and the ESFAS function setpoints were originally calculated by Westinghouse, using the Westinghouse methodology described in WCAP-12523, which was approved by the staff. For the uprated power operation, an assessment of setpoints and allowable values for the RTS and the ESFAS functions was performed using the same Westinghouse methodology, and the results indicated that no RTS or ESFAS setpoints, and allowable values need to be changed for the proposed power uprate.

For the other safety-related and the BOP instrument functions, setpoints and allowable values for the proposed power uprate were established using the licensee's in-house "ComEd [now Exelon] setpoint methodology." The licensee stated that the ComEd setpoint methodology is based on guidelines of ANSI/ISA S67.04, Parts 1 and 2, 1994, and was reviewed and approved by the staff as part of Instrumentation and Control (I&C) inspection at Dresden Nuclear Power Station in 1994. An evaluation of the ComEd setpoint methodology was included in Amendment No. 129 to Facility Operating License No. NPF-11 for LaSalle County Station. The staff in approving the amendment, concluded that the methodology addresses the proper terms for establishing setpoints. The licensee stated that the methodology is based on conservative licensing analyses or conservative design, operating limits, plant operating experience and establishes instrument uncertainties at 95 percent probability and a 95 percent confidence level. Apart from the exceptions noted in the plants' UFSAR, this methodology does not deviate from guidance provided in RGuide 1.105, Revision 1.

4.15.1 NSSS and BOP Control Systems

The licensee has performed a detailed evaluation of each affected NSSS and BOP control system to determine the impact of the power uprate conditions. As a result of the evaluation, the reactor coolant average temperature program will be modified to maintain the desired programmed reactor coolant temperature and instruments for RCS ΔT and T_{ave} will be rescaled. The ΔT indication will be renormalized to the uprated calorimetric power indication, and the programmed RCS temperature control will be adjusted at a new full-power-T_{ave}. The OTΔT penalty and penalty bands will be re-scaled. In addition, the following control systems need minor adjustment to accommodate changes to RCS operating temperature: the pressurizer level program (to reflect change in water expansion from zero to full power), the high T_{ave} alarm setpoint, and the steam dump control system.

4.15.2 Suitability of Existing Instruments

In its submittal, the licensee stated that for the proposed power uprate, each existing instrument of the affected NSSS and BOP systems was evaluated to determine its suitability for the revised operating-range of the affected process parameters. Where operation at power uprated condition impacted safety analysis limits, the evaluation verified that the acceptable safety margin continued to exist under all conditions of the power uprate. Where necessary, setpoint and uncertainty calculations for the affected instruments were revised. The licensee's evaluation to determine instrument suitability identified the following cases:

- Existing instrumentation was found to be adequate to accurately measure the range and the normal operating point of the process variables, but the existing calibrated range does not envelop the power-uprated conditions. In this case, affected instruments will be re-calibrated and setpoints will be readjusted.

- Existing instrumentation was found not to be adequate to accurately measure the range and the normal operating point of the process variables, because the instrument cannot be calibrated to envelop the revised range of the affected process variable. These instruments will be replaced with the suitable ones and will be recalibrated and their setpoints will be readjusted.

- An existing instrument was found to be scalable, but the branded operating-region needed to be changed for the revised operating band and/or the operating point. In this case, the meter scale will be replaced and instruments will be recalibrated to envelop the revised operating range.

Apart from a few devices that needed change, the licensee's evaluations found most of the existing instrumentation acceptable for the proposed power uprate operation. As a result of the evaluation the following changes were identified:

- The range of Byron, Unit 1 high pressure turbine first stage pressure transmitters was found to be unsuitable. Therefore, these transmitters will be replaced.

- The Byron and Braidwood Unit 2 turbine first-stage pressure transmitters will be rescaled because, first-stage impulse pressure will be decreased at these plants as a result of uprate.

- The steam generator (SG) narrow-range level transmitters will be rescaled to reflect change in water density for accurate water level indication.

- RCS ΔT and T_{ave} will be rescaled. Setpoints for low-temperature overpressure protection will be revised based on the Westinghouse evaluation of reactor vessel fluence.

- Pressure relief tank (PRT) high and low level alarms will be adjusted in accordance with the Westinghouse revised setpoint analysis. The licensee stated that the revised PRT setpoints provide more operating margin.

- The Byron and Braidwood Unit 2 alarm for steam line pressure will be adjusted for the lower steam pressure.

- SG feedwater (FW) flow high alarm will be adjusted for the increased FW flow.

- The setpoints of several radiation monitors will be adjusted for the uprated power conditions. In addition, the alarm setpoint for the following will be changed:

  + the insertion limit alarm,
  + the high auctioneered T_{avg} temperature alarm,
  + the low steam line pressure alarm,
  + containment atmosphere monitoring alarms,
  + grossly failed fuel process radiation monitoring alarms, and
  + the control room air intake radiation alarm.

- As a result of evaluation, the licensee lowered the containment high-pressure setpoint slightly, reducing the margin between the safety analysis limit and the trip setpoint. However, the licensee concluded that the existing containment pressure transmitters were acceptable for the power uprate conditions because a positive margin has still been maintained.

- A reduction in margin was also noted when the main steam line low-pressure calculations were reviewed to address the increased temperature and the revised safety analysis limit; in this case also, the licensee had found the existing steam line pressure transmitters acceptable because of a positive margin.

- The existing settings of the FW pump net positive suction head protection circuit will be revised.

- The FW pump speed control instrument scaling calculations will be revised, and instrument scaling will be adjusted to implement the revised RCS temperature parameters for T_{hot}, T_{avg} and T_{cold} functions.

- Plant process computers will need rescaling or setpoint modifications only for analog inputs from those instrumentation whose scaling or setpoints have been changed for the power uprate. The plant simulators will also be modified to replicate the revised simulation control panel hardware and software changed for the affected setpoints and system components.

The above-described changes will be performed to accommodate the revised process parameters, and the staff agrees with the licensee's conclusion that, with the above noted minor modifications and changes, the Byron and Braidwood instrumentation and control systems will accommodate the proposed power uprate without compromising safety.

4.16 Reactor Trip Time Delays

There are various instrumentation delays associated with each reactor trip function that are modeled directly and considered in the non-LOCA safety analyses. The total delay time is defined as the time from when the trip conditions are reached to the time the rods are free to fall. The safety analysis trip setpoint and maximum time delay assumed for reach reactor trip function were provided in on page 6-106 of the licensee's July 5, 2000, submittal. The licensee stated that the values are the same as those applicable to the current licensing basis non-LOCA safety analyses and remain applicable for the power uprate. The staff did not perform a further review of the reactor trip time delays.

5.0 BOP EVALUATION

5.0.1 BOP Piping

The licensee evaluated the adequacy of the BOP piping systems based on comparing the existing design bases parameters with the core power uprate conditions. The BOP piping systems evaluated for the power uprate are main steam, feedwater, SG blowdown, auxiliary feedwater, extraction steam, heater drains, condensate, turbine plant cooling, secondary sampling, spent fuel pool cooling, residual heat removal, component cooling, and essential service water. The evaluation was performed by conservatively scaling up piping stresses and loads using the ratio of the power uprate temperature, pressure and flow rate conditions to the corresponding pre-uprate operating conditions. The results of the evaluation are summarized in Section 9.3.20 of the licensee's July 5, 2000, submittal. In general, all BOP piping systems affected by the power uprate have a scaling factor less than 1.08 which is determined by the licensee to be within the allowable limits. The staff finds the methodology to be acceptable considering the conservatism in the calculation of the scaling factors for the power uprate stresses and loads. The licensee concluded that all piping systems at Braidwood and Byron stations remain acceptable and will continue to satisfy existing design-basis requirements under uprate conditions in accordance with the ASME Section III 1974 Edition up to Summer 1975 Addenda and American National Standards Institute (ANSI) B31.1, 1973 Edition, which is the Code of record. The staff agrees with the licensee's conclusion considering the power uprate impact to be insignificant in combination with other loading such as seismic.

In addition, the design bases high energy pipe break (HEPB) analyses were also reviewed by the licensee to evaluate the effects of the uprate conditions on the pipe break locations, jet thrust and jet impingement forces, which were used in the plant hazard analyses, and the design of pipe whip restraints. The review verified that the existing postulated pipe break locations are not affected since the design bases piping analyses will not change due to the power uprate. The current design bases for jet thrust and jet impingement forces due to postulated pipe breaks for these systems are not affected by the uprate, since the systems do not experience a pressure increase as a result of the core power uprate. On the basis of its review, the staff concurs with the licensee's conclusion that the original design analyses for the pipe break locations, jet thrust, jet impingement, and pipe whip restraints, are unaffected by the power uprate.

5.1 Main Steam System

The licensee performed evaluations of the effects resulting from plant operations at the proposed uprated power level on the main steam system including the MSIVs, steam generator power operated relief valves (PORVs), MSIV bypass valves, and main steam safety valves (MSSVs). The licensee stated that plant operations at the proposed uprated power level will increase the steam mass flow by approximately 5.8 percent for Byron and Braidwood Unit 1 and approximately 5.4 percent for Byron and Braidwood Unit 2. The above existing components are adequately sized for the uprating conditions. Therefore, the licensee concluded that plant operations at the proposed uprated power level will have an insignificant or no impact on the main steam system and its associated components.

Based on the staff's review and the experience gained from staff review of power uprate applications for similar PWR plants, the staff concurs with the licensee that plant operations at the proposed uprated power level will have an insignificant or no impact on the main steam system and its associated components.

5.2 Miscellaneous Main Steam Auxiliary Systems

5.2.1 Steam Dump System

The licensee evaluated the steam dump system for the plant operations at proposed uprated power level, and concluded that there will be an insignificant or no impact on the steam dump system. Based on the experience gained from staff review of power uprate applications for similar PWR plants, the staff concurs with the license that operation of the steam dump system at the proposed uprated power level is acceptable.

5.2.2 Heater Drain System

The heater drain system (HDS) is a non-safety related system that collects condensed steam from the feedwater heater, drain coolers, reheaters, and moisture separators. The licensee evaluated the HDS for plant operations at the proposed uprated power level and concluded that they will have an insignificant or no impact on the HDS. Since this low pressure heater drain system does not perform any safety related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on its design and performance.

5.2.3 Extraction Steam System

The extraction steam system is designed to provide steam at various pressures and temperatures to preheat condensate and feedwater as it flows from the main condensers to the steam generators. Since the extraction steam system does not perform any safety related function, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the extraction steam system.

5.3 Condensate and Feedwater System

The licensee performed evaluations of the effects of Byron and Braidwood plant operations at the proposed reactor power level on the condensate and feedwater systems. The licensee concluded that the existing condensate and feedwater systems at Byron and Braidwood are adequate for power uprate conditions.

Since the condensate and feedwater systems do not perform any safety related function and their failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of Byron and Braidwood plant operations at the proposed reactor power level on the design and performance of this systems.

5.4 Circulating Water System

The circulating water system is designed to remove the heat rejected to the condenser by turbine exhaust and other exhausts over the full range of operating loads, thereby maintaining adequately low condenser pressure. The licensee stated that performance of this system was evaluated for power uprate and determined that the system is adequate for uprated power level operation.

Since the circulating water system does not perform any safety function and its failure will not affect the performance of any safety-related system or component, the staff has not reviewed the impact of plant operations at the proposed uprated power level on the designs and performances of this system.

5.5 Service Water System

The service water system (SWS) which consists of the essential service water system (ESWS) and the non-essential service water system (NESWS) is designed to provide reliable supplies of cooling water to various safety-related and non-safety related equipment during normal plant operations, a station blackout event and following a design basis accident.

5.5.1 ESWS

The ESWS is designed to supply cooling water to various safety-related systems and other essential equipment during normal plant operations, a station blackout event, a LOCA, or main steamline break accident. The licensee performed evaluations and stated that the ESWS as designed will supply sufficient water to remove the additional heat loads resulting from plant operations at the proposed uprated power level.

Based on staff review and the experience gained from its review of power uprate applications for similar PWR plants, the staff finds that plant operations at the proposed reactor uprated power level do not change the design aspects and operations of the ESWS. Therefore, the

staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the ESWS.

5.5.2 NESWS

The NESWS is designed to supply cooling water to various non-safety related components and heat exchangers in the turbine, reactor, and radwaste buildings during normal plant operation. The licensee performed evaluations of the effects of these increases in heat loads on NESWS and stated that the NESWS has the capacity to accommodate the additional heat loads.

Since the NESWS does not perform any safety function and its' failure will not affect the performance of any safety-related system or component, the staff did not review the impact of plant operations at the proposed uprated power level on the designs and performances of this system.

5.5.3 Ultimate Heat Sink

The ultimate heat sink (UHS) provides a heat sink for processing and operating heat loads from safety related components during a transient or accident. In addition, the UHS provides the safety-related source of auxiliary feedwater when the condensate storage tank (CST) is not available.

At Byron Station, the UHS is composed of two mechanical-draft cooling towers and the make-up system to these towers.

The Braidwood Station UHS consists of an excavated essential cooling lake integral with the main Braidwood cooling lake.

The licensee performed evaluations and concluded that the UHSs for both Byron and Braidwood will provide sufficient cooling water under a design-basis accident for plant operations at the proposed uprated power level.

Based on staff review and the experience gained from its review of power uprate applications for similar PWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the UHS.

5.7 Main Generator and Auxiliaries

The licensee performed evaluations to determine the maximum generator reactive power (MVAR) output at the worst-case bounding conditions, to determine if the calculated heat load to the generator coolers would exceed the heat load based on the generator nameplate. The licensee also evaluated the adequacy and margin of the present generator coolers (hydrogen cooling equipment and stator water cooler) for operation under power uprate conditions. The licensee states that the "worst case/bounding condition" has been determined to occur in the summer months, with the unit operating at an uprated power of 1247 MW, generator voltage of 26 kV (not the rated 25 kV), and holding the generator heat load (MW) constant while varying the MVAR output in various increments. The maximum generator MVAR limit under this condition has been established at 530 MVAR, which results in machine operation at less than the nameplate output rating (1361 MVA). Also, the calculated heat load to the generator

coolers from operating under the power uprate conditions will not exceed the heat load for the generator nameplate rating. Since the anticipated power output levels will not exceed the main unit generator's name plate output rating, the staff concludes that the main unit generator is capable of operating satisfactory at the power uprate.

5.8 Heating, Ventilation, and Air Conditioning System

The licensee evaluated the following heating, ventilation, and air conditioning (HVAC) systems to ensure that margin and capability exist to operate satisfactorily to support the plant thermal power uprate from 3525 MWt to 3,600.6 MWt (including 14 MWt from reactor coolant pump heat):

- Control Room HVAC System
- Spent Fuel Pool Area Ventilation System
- Auxiliary Building and radwaste Area Ventilation Systems
- Turbine Area Ventilation System
- Engineered Safety Features Ventilation Systems
- Pump House Ventilation Systems
- Off-gas Miscellaneous Tank Vent Filter System
- Containment Ventilation System
- Primary Containment Purge System
- Miscellaneous HVAC System

The licensee indicated that the impact of power uprate on the HVAC systems listed above is the increase in the amount of component heat loss to the environment. This increase is in proportion to the new electrical load for motors and other equipment (i.e. electrical and control panels, cables, etc) or increases in the operating temperature of piping and other hot fluid containing components.

The licensee evaluated the performance of the above systems by performing an analysis to determined the impact of power uprate on the HVAC systems ability to maintain operating temperatures. The licensee concluded that the power uprate does not impact the plant HVAC systems ability to maintain the operating environment temperature at or below the respective normal operating temperatures.

Based on the review of the licensee's rational and the experience gained from staff review of power uprate applications for similar PWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed power uprated will have an insignificant impact on the HVAC systems.

5.9 Radwaste Systems (Solid, Liquid, and Gaseous)

The solid, liquid, and gaseous radwaste activity is a function of the reactor core power. The licensee evaluated the existing design of the radwaste systems and concluded that plant operations at the proposed uprated power level do not change the design aspects and operations of the radwaste systems and will have an insignificant impact on the radwaste systems.

The waste processing system processes liquid and gaseous wastes from the plant. The sources of liquid radioactive wastes consist mostly of the leaking reactor coolant from piping and equipment. Increasing the rated core thermal power will result in a slight increase in the concentrations of RCS isotopic decay products, and thus in slightly higher activities of the leaking reactor coolant. However, the difference will be too small to affect the way liquid wastes are treated.

Generation of gaseous effluents, consisting of fission gases, is proportional to reactor power. However, only a very small fraction of these gases ever reaches the waste gas system and the increase of activity after power uprate will be minimal. In some cases it may require an extended holdup time, but in most cases a normal holdup time of 45 to 60 days will bound the required holdup time for increased activities. The licensee concluded that there will be no need for modifying the waste processing system. The staff concurs with this conclusion.

5.10 Additional BOP Reviews

The impact of plant operations at the proposed power uprate power level on HELB outside containment and equipment environmental qualification is addressed in Sections 4.12 and 5.0.1 of this SE.

5.10.1 Miscellaneous Systems Not Impacted by Power Uprate

The licensee stated that various systems (i.e. auxiliary steam, condenser offgas, chilled water, emergency diesel generators and their auxiliaries, etc.) were evaluated and found to be not affected or insignificantly affected by plant operations at the proposed uprated power level. These systems were evaluated for respective capacities, heat removal capabilities, and in many cases no direct connection to plant power uprate was found.

Since plant operations at the proposed uprated power level do not change the design aspects and operations of these systems, and from the experience gained from staff review of power uprate applications for other PWR power plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant or no impact on the these systems.

5.11 Miscellaneous Electrical Reviews

Increased generator output (MWe) results not only in an increase in generator output power delivered to the transmission grid, but also a change in the required brake horsepower over kW (BHP/kW) loading for some large auxiliary loads. It is therefore necessary to establish that the electrical distribution systems and components have the capacity to carry increased current and that the loads will operate satisfactory at the power uprate conditions while the transmission system grid remains stable. The staff reviewed the primary electrical distribution systems to establish the impact of the increased main unit generator power output under power uprate conditions on the electrical systems and components.

5.11.1 Main Power Transformers (MPTs)

The licensee reviewed the existing sizing calculation for the MPTs to confirm that they have sufficient capacity and margin to handle the electrical power requirements under power uprate conditions. The licensee also evaluated the adequacy of the present MPT cooling system for

operation under the power uprate conditions. The licensee has determined that the existing MPTs have sufficient capacity and margin to support the output of the main unit generator at power uprate conditions. Since the power output at the uprated level is still within the MPT rating, the staff concludes that the MPT has sufficient capacity and margin to handle the electrical power requirements under power uprate conditions.

5.11.2 System Auxiliary Transformers (SATs)

The licensee has determined that the existing SATs have sufficient capacity and margin to support operation at power uprate conditions without modification. Since the power uprate output still will be within the SAT rating, the existing SAT cooling design is also adequate for the power uprate. The licensee also confirmed that bus voltage is essentially unchanged at power uprate loading conditions. Accordingly, plant operation at power uprate conditions will have no effect on loss of voltage or degraded grid voltage protection schemes and motor starting scenarios. In addition, the licensee performed short circuit analysis and confirmed that short circuit values are essentially unchanged at power uprate loading conditions. Since the power uprate output is still within the SAT rating, the staff concludes that the SATs remain adequate for operation at the higher power levels.

5.11.3 Unit Auxiliary Transformers (UATs)

The licensee performed a review to confirm that the UAT at each unit has sufficient capacity and margin to handle the electrical power requirements under the power uprate conditions. The licensee also evaluated the adequacy of the present UAT cooling system for operation under power uprate conditions. The licensee has determined that the existing UATs have sufficient capacity and margin to support operation at power uprate conditions without modification. Since the power uprate output is still within the UAT rating, the staff concludes that the UATs have sufficient capacity and margin to handle the electrical requirements under the power uprate conditions.

5.11.4 Isolated Phase Bus Ducts

The licensee evaluated the rated capacity of the main generator isolated bus duct connection to the MPT and the isolated phase bus duct taps for the UATs for capacity and margin under power uprate conditions. The licensee also evaluated the adequacy and margin of the present isolated phase bus duct cooling equipment for operation under power uprate conditions and determined that the existing isolated phase bus ducts have sufficient capacity and margin to support the output of the main unit generator at power uprated conditions. As the design rating of the isolated phase bus duct is not exceeded, the existing cooling design is considered adequate for the power uprate. Since the existing isolated phase bus ducts have sufficient capacity, the staff concludes that isolation phase bus ducts will support the generator output at the uprated conditions.

5.11.5 Emergency Diesel Generators (EDGs)

For the power uprate conditions, the licensee evaluated the engineered safety feature (ESF) bus loading with a concurrent loss of power (LOOP) and LOCA to determine if; (1) it was within the design ratings of the diesel generators, and (2) the diesel generators would remain capable of performing their safety-related functions. The licensee determined that the present diesel

generator loading analysis bounds the power uprate diesel generator loading. Since the present diesel generator loading analysis bounds the power uprate diesel generator loading, the staff concludes that the diesel generator will not be impacted by power uprate.

5.11.6 Nonsegregated Bus

The licensee compared the rated capacities of the nonsegregated phase bus ducts, which connect the UATs and SATs and their respective switchgear, to the anticipated load of the associated switchgear under power uprate. The licensee determined that the existing nonsegregated phase bus ducts have sufficient capacity and margin to support operation at power uprated conditions without modification. Since the existing nonsegregated phase bus ducts have sufficient capacity and margin to support operation at power uprated conditions, the staff concludes that the nonsegregated bus will remain adequate for operation at the higher output.

5.11.7 Large Loads and Cables

The licensee performed system evaluations to determine the anticipated effect of the power uprate conditions on the large medium-voltage loads for the plant startup conditions, normal operating conditions, and LOCA conditions. Where a load increase (BHP/kW) was identified, its impact on the equipment performance and associated cable ampacity was evaluated. Accelerated aging and reduction in design life were also considered when the motor might be required to operate at a load exceeding its nameplate rating (i.e., the reactor coolant pump during cold loop operation). The licensee determined that some of the large medium-voltage motors experience a BHP/kW change (increase or decrease) at power uprate conditions. However, except for the RCP, the BHP remains within the nameplate rating of the motors. The ampacity of the motor cables remains adequate, since the cable sizing is typically based on equipment nameplate ratings.

The cold loop rating of the RCP at power uprate exceeds the nameplate cold loop rating of the motor. However, the licensee states that its analysis indicates that this increase will reduce the design life of the motor by approximately one month. The ampacity of the RCP motor cables remains adequate. The staff concludes that a reduction in the RCP design life of one month is acceptable because the RCPs are not operating in the cold loop operation for an extended period of time and one month is not an significant period of time over the design life of RCPs. Therefore, the large auxiliary loads will continue to satisfactorily perform their intended function.

5.11.8 Protective Relay Settings

The licensee evaluated the impact of the power uprate on station protective relay schemes and setpoints. The licensee stated that the existing station protective schemes and setpoints will not be affected by operation under the power uprate conditions because the data upon which protective relay settings are typically based (equipment nameplate ratings, motor and cable thermal data, and short circuit studies) are essentially not affected by power uprate conditions. Since the existing station protective schemes and setpoints are not affected by the power uprate, the staff concludes that there is no impact of the power uprate on the station protective relay schemes.

5.11.9 Grid Stability

The staff reviewed the licensee's evaluation of grid stability to determine if the Byron and Braidwood stations will continue to be in full compliance with the General Design Criterion (GDC) 17, "Electric Power System," as a result of the power uprate. The licensee performed dynamic and transient stability analyses for each station to study stability issues for operation under the power uprate conditions. The criteria used are in accordance with the Mid-American Interpool Network (MAIN) Guide No. 2, as stated in Section 8.2.2 of the Byron and Braidwood UFSAR. The results of these studies were used to evaluate the impact of power uprate on transmission system grid stability under normal expected operating conditions for double line contingency events and faults. The licensee also used the studies to determine operating limitations, and proposed modifications to resolve the stability issues. Based on these studies, the licensee has identified new modifications (including unit trip schemes, a reduction of the existing local breaker backup timer settings, and installation of a power system stabilizer (PSS) on Byron, Unit 2) that would be required to maintain stability in the transmission grid. The licensee also has completed dynamic and transient stability analyses for Braidwood, Units 1 and 2. The licensee has identified a reduction of the existing local breaker backup (LBB) timer settings required to maintain stability in the transmission system grid. The licensee stated that it would implement these modifications prior to power uprate to maintain the stability of the Byron and Braidwood transmission grid. Modifications for Byron Station include revising the unit trip schemes, lowering the existing LBB timer settings, and installing a PSS on Byron Unit 2. Braidwood Station modification will include a reduction of the existing LBB timer settings. With the licensee's proposed modifications, the staff concludes that the Byron and Braidwood stations will continue to be in compliance with GDC 17 and the power uprate will not adversely affect the stability of the transmission system grid.

6.0 <u>HUMAN FACTORS</u>

The staff reviewed the licensee's submittal dated July 5, 2000, as well as its November 27, 2000, January 31, 2001, and February 28, 2001, responses to the staff's RAI of October 19, 2000. Following is the staff's SE input which is based on five review topics.

Topic 1 - *Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?*

The licensee stated in its letter dated July 5, 2000 (page A-21) that, "The Power uprate has the potential to affect plant procedures used to operate and maintain the facility in accordance with design basis and licensing requirements... . Procedures that are identified as being affected by the power uprate will be revised prior to the uprate implementation." In Attachment E (page 12-1), the licensee stated that, "A physical review of each procedure identified [,during the screening,] will be conducted to determine the need for revision. Those procedures will be revised to incorporate changes. For example, changes due to modifications, operator response times, setpoint changes will result in revisions to existing procedures... and all required training will be conducted prior to the implementation of the power uprate."

In addition to the responses provided by the licensee in its July 5, 2000, submittal, the staff, in its October 19, 2000 RAI, asked the licensee to identify what specific procedures will be changed, what changes will be made to the procedures and, what new operator actions will be required.

In its November 27, 2000, response to the staff's RAI, the licensee stated that the major effect of the power uprate will be on Appendix J surveillance procedures. These procedures will be revised to reflect changes in the Technical Specifications (e.g., post- accident peak containment pressure). The proposed power uprate will effect one operator action response time (further addressed in the response to Topic 2, following). The proposed power uprate will cause range changes to process parameters (e.g., pressures, temperatures, and flows), that will result in "several procedures being changed to reflect the uprated parameters."

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the type and scope of plant emergency and abnormal procedures that will be affected by the uprate, has indicated that the procedures will be appropriately revised, operators will be trained on the changes before the procedures are implemented, and has adequately described the effect of the procedure changes on operator actions.

Topic 2 - *Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (either reduced or increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator exercises conducted to assure that operator response times for operator actions that are potentially sensitive to power uprate can be successfully achieved within allowable time limits.*

The licensee stated in its November 27, 2000, response to the staff's RAI, that, "Results of the accident analysis showed that required operator response times are not affected by power uprate; however, as part of the activities performed in support of power uprate, the assumed operator response time to isolate auxiliary feedwater to the faulted steam generator during a feedwater line break event was administratively reduced from 30 minutes to 20 minutes. This change was made to establish consistency with the assumed operator response time for the steamline break outside containment. Both of these events direct the operators to the same emergency response procedures; therefore, making the response time consistent is beneficial from a human factors standpoint."

In a supplemental response to the staff's RAI submitted by the licensee, January 31, 2001, and February 28, 2001, the licensee clarified that the operator actions required to isolate auxiliary feedwater (AFW) to the faulted steam generator(SG) during a <u>feedwater line break event</u> and the operator actions required for the <u>steamline break outside containment are identical</u>; the same emergency operating procedure, "Faulted Steam Generator Isolation Unit 1(2) [EP-2]," is used by operators to mitigate both events. The licensee indicated that the operators' ability to complete the faulted steam generator isolation for the steamline break event had been previously demonstrated by personnel from 13 different operating crews on the training simulator. "All crews completed isolation of the AFW to the faulted SG during a MSLB using procedure EP-2 in less than 20 minutes. The average crew time to isolate the faulted SG was approximately 7.4 minutes while the maximum time was 17 minutes."

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified operator actions sensitive to the power uprate, described the effect of the power

uprate on these actions, and adequately explained how the changes to the operator actions have been validated.

Topic 3 - *Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.*

In Attachment C (page C-24) of the licensee's July 5, 2000, submittal, the licensee indicates that, "The basic design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any reactor trip or ESF actuation setpoints." However, the licensee also state that, "Minor modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components."

In addition to the information provided by the licensee in its July 5, 2000, submittal, the staff, in its October 19, 2000, RAI, asked the licensee to describe "the "minor modifications" referred to by the licensee and what effect they will have on operator performance.

In its November 27, 2000, response to the staff's RAI, the licensee stated that instrumentation setpoint/scaling adjustments are being performed within the 7300 control system at both Byron and Braidwood stations. Adjustments to the new Tavg and ΔT are being incorporated. Adjustments to the turbine driven feedwater pumps speed control systems will also be performed. The licensee also indicated that any annunciator alarm points or computer points that are affected by the setpoint or scaling adjustments will be adjusted appropriately. The effect that any adjustments to computer points might have on station displays will also be assessed. All meter scale adjustments not covered under an already identified instrument setpoint or scaling adjustment, will be performed under the stations' Design Change Process. All zone banding changes will be documented and submitted for plant review, with all changes being made in accordance with the ComEd Human Factors Manual and appropriate station procedures.

The licensee stated that the "minor changes" being performed at Byron and Braidwood Stations include setpoint and scaling adjustments to several instruments and controls that are further described in another section of this SE. Mechanical component changes involve valve replacements and valve trim changes to increase the valve flow coefficient for several drain valves and the addition of gland steam piping to increase the high pressure turbine gland leak-off removal capacity.

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the changes that will occur to alarms, displays, and controls, as a result of the power uprate and adequately described how these changes will be accommodated.

Topic 4 - *Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.*

In its November 27, 2000, response to the staff's RAI, the licensee stated that, "No changes will be made to the process parameters that input to SPDS and no physical changes will be made to the SPDS display from a human factors perspective."

The licensee further stated that, "process computer points are being reviewed and updated as required to support implementation of the power uprate." The field data computer points associated with the SPDS will be re-scaled as necessary and needed software changes will be made before the power uprate is implemented. The licensee's design change process will address all meter scale adjustments not covered under an already identified instrument setpoint/scaling adjustment. The licensee's Operations and Training Departments will assess the need for revising the operator training program, make required changes and conduct the appropriate training.

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the changes that will occur to the SPDS as a result of the power uprate and adequately described how the changes will be addressed.

Topic 5 - *Describe all changes the power uprate will have on the operator training program and the plant simulator.*

In its November 27, 2000, response, the licensee stated that appropriate lesson plans will be revised to reflect the new operating conditions resulting from the power uprate and that operators will be trained on expected system parameter changes and on the effects of the uprate on BOP and NSSS margins. Just prior to the actual uprates, operators will receive "Heightened Level of Awareness" training on the uprate program.

Changes to the simulator, of which the licensee provided an initial list, will be implemented before the uprate according to the power uprate implementation schedule.

A supplemental response (January 31, 2001) to the staff's RAI, submitted by the licensee clarified that simulator modifications will be made in accordance with the ANS/ANSI 3.5, "Nuclear Power Plant Simulators for Use in Operator Training" standard. For the power uprate, the station will provide initial training to the licensed operators before the on-line uprate using data from applicable power uprate calculations. Operators will be informed that they are receiving training based on project calculations. After the power uprate, plant data will be collected and compared to the simulator data to ensure that the simulator meets the performance criteria specified in ANS/ANSI 3.5, Section 4.

In its February 28, 2001, submittal, the licensee clarified that required simulator modifications will be made in accordance with the 1985 revision to the ANS/ANSI 3.5, "Nuclear Power Plant Simulators for Use in Operator Training" standard, Section 5.3, "Simulator Modifications," and Section 5.4, "Simulator Testing."

The staff finds the licensee's response satisfactory because the licensee has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes in accordance with the requirements of ANS/ANSI Standard 3.5.

The staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

7.0 EVALUATION OF CHANGES TO FACILITY OPERATING LICENSES AND TECHNICAL SPECIFICATIONS

7.1 Evaluation of Changes to the Operating Licenses and TSs

7.1.1 License Condition 2.C.(1) and Rated Thermal Power Definition

License Condition 2.C.(1) and the definition of rated thermal power (RTP) is being revised from 3411 MWt to 3586.6 MWt. The license has provided the results of its reanalyses or evaluation including LOCA and non-LOCA transients and accidents, containment response, radiological consequences, NSSS, and BOP systems and components to support the operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, at the uprated power level. The staff has reviewed the licensee's submittal and concludes that each unit can safely operate at a core power of 3586.6 MWt.

7.1.2 Dose Equivalent I-131 Definition

The definition was changed to add RG 1.109 and International Commission on Radiation Publication 30 (ICRP 30) as references. The two references incorporate recent information gained in the areas of radionuclide dosimetry and biological transport in humans. The staff finds the additions to be acceptable.

7.1.3 Change in DNBR Acceptance (TS 2.1.1.1)

TS 2.1.1.1, "Reactor Core Safety Limits," currently states, "In Mode 1, the DNBR shall be maintained ≥ 1.25 for the WRB-2 DNB correlation." This requirement is currently applicable for both a thimble cell and a typical cell. This safety limit would be changed to require the DNBR to be maintained ≥ 1.24 for the WRB-2 DNB correlation for a thimble cell and ≥ 1.25 for the WRB-2 DNB correlation for a typical cell. The staff has reviewed the effects of this proposed change on the safety analyses for uprated conditions and concludes that there will be at least a 95 percent probability that DNB will not occur on the limiting fuel rods during normal operation, operational transients, and AOOs, at the 95 percent confidence level. Therefore, fuel damage is not expected to occur for these conditions and the proposed TS change is acceptable.

7.1.4 Change in RCS Total Flow Rate (TS 3.4.1)

TS 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," would be changed to increase the minimum RCS total flow rate from ≥ 371,400 gpm to ≥ 380,900 gpm. Surveillance Requirements 3.4.1.3 and 3.4.1.4 would be revised accordingly. The staff has reviewed the power uprate analyses, which assumed a total RCS flow rate value of 368,000 gpm for all normal and accident conditions. The proposed TS value of 380,900 gpm conservatively bounds the analysis value and accounts for flow measurement uncertainty and maximum SG tube plugging level. The analyses have shown that the acceptance criteria for all normal and accident conditions continue to be met and we conclude that the proposed TS value of 380,900 gpm conservatively bounds the analyses value and is, therefore, acceptable.

7.1.5 Steam Generator Plugging or Repair Limit (TS 5.5.9.e.6)

TS 5.5.9.e.6 currently permits a plugging limit of 40 percent of nominal wall thickness for laser welded sleeves. The proposed change would change the plugging limit to 38.7 percent. The licensee performed sleeve wall-thinning calculations for degraded sleeving in accordance with RG 1.121, which specifies that the sleeve should maintain a safety margin of three under the primary-to-secondary pressure differential under normal operating conditions. The licensee's calculations showed that the plugging limit of 38.7 percent for laser welded sleeves is conservative under the pressure loading of 3ΔP in the power uprate condition. Incorporation of the change into the proposed TS change is acceptable.

7.1.6 Containment Leakage and Rate Testing Program (TS 5.5.16)

The licensee has proposed to revise the peak calculated containment internal pressure for the design basis LOCA from 47.8 psig to 42.8 psig for Units 1, and from 44.4 psig to 38.4 psig for Units 2. The licensee calculated the mass and energy releases which incorporate revised design parameters corresponding to the new power levels. The staff reviewed the licensee's evaluation and concurs with the proposed change.

8.0 STATE CONSULTATION

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

9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and find no significant impact was published in the Federal Register on January 5, 2001 (66 FR 1158).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

10.0 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) 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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Tables

TABLE 1

Radiological Consequences (rem)
Byron Station, Units 1 and 2

Design Basis Accidents	EAB Thyroid	EAB WB[1]	LPZ Thyroid	LPZ WB	Control Room Thyroid	Control Room WB
LOCA	61	3	7	<1	15	<1
MSLB						
Pre-accident	4.6	<1	0.5	<1	29	<1
Accident-initiated	5.0	<1	2.0	<1	13	<1
SGTR						
Pre-accident	11	<1	3.6	<1	3.3	<1
Accident-initiated	9.0	<1	0.3	<1	0.3	<1
FHA	56	3.8	1.7	<1	1.3	<1
Locked rotor	13	<1	1.0	<1	16	<1
Rod ejection	34	1	6	<1	27	<1

[1] Whole body

TABLE 2

Radiological Consequences (rem)
Braidwood Station, Units 1 and 2

Design Basis Accidents	EAB		LPZ		Control Room	
	Thyroid	WB	Thyroid	WB	Thyroid	WB
LOCA	82	4	34	<1	15	<1
MSLB						
Pre-accident	6.2	<1	2.6	<1	13	<1
Accident-initiated	3.6	<1	4.8	<1	29	<1
SGTR						
Pre-accident	14	<1	1.3	<1	1.1	<1
Accident-initiated	12	<1	1.1	<1	1.0	<1
FHA	75	5.1	7.0	<1	1.4	<1
Locked rotor	18	<1	2.0	<1	16	<1
Rod ejection	45	<1	22	<1	26	<1

Table 3
Parameters and Assumptions Used in Radiological Consequence Calculations Loss-of-Coolant Accident

Parameter	Value
Reactor power	3658.3 MWt
Containment volume of sprayed region	2.35E+6 ft^3
Containment volume of unsprayed region	5.0E+5 ft^3
Flow rate from sprayed to unsprayed region	1.06E+5 cfm
Flow rate from unsprayed to sprayed region	1.06E+5 cfm
Containment leak rate to environment	
0 - 24 hours	0.1 percent per day
1 - 30 days	0.05 percent per day
Spray removal rates	
Elemental iodine	20 per hour
Time to reach DF[1] of 100	Not reached
Particulate iodine	6 per hour
Time to reach DF[1] of 50	Not reached
Spray operation	
Initiation time	90 seconds
Termination time for injection	22.4 minutes
ECCS leak rate	
0 to 30 days	7820 cc/hr
Iodine partition factor	10 percent
Sump volume	38979 ft^3
Auxiliary building exhaust filter efficiency	90 percent
Auxiliary building exhaust filter bypass	1 percent
Control room isolation time	15 seconds

[1] Decontamination factor

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Main Steamline Break Accident

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity prior to accident	
Pre-existing spike	60 μCi/gm DEI-131 42.6 μCi/gm I-131
Accident-initiated spike	1.0 μCi/gm DEI-131 0.77 μCi/gm I-131
Secondary coolant iodine activity prior to accident	0.1 μCi/gm DEI-131 0.077 μCi/gm I-131
Iodine spike (appearance) rate increase	500 times
Duration of accident-initiated spike	6 hours
Steam generator tube leak rates	
Faulted steam generator	0.5 gpm
Intact steam generators	0.654 gpm from 3 steam generators
Steam releases	
Faulted steam generator	1.67E+5 lbs
Intact steam generators	
0 to 2 hours	4.42E+5 lbs
2 to 8 hours	9.77E+5 lbs
8 to 40 hours	2.216E+6 lbs
Duration of activity release	40 hours
Control room isolation time	5 minutes

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Steam Generator Tube Rupture Accident

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity prior to accident	
Pre-existing spike	60 µCi/gm DEI-131 42.6 µCi/gm I-131
Accident-initiated spike	1.0 µCi/gm DEI-131 0.77 µCi/gm I-131
Secondary coolant iodine activity prior to accident	0.1 µCi/gm DEI-131 0.077 µCi/gm I-131
Iodine spike (appearance) rate increase	500 times
Duration of accident-initiated spike	8 hours
Steam generator tube leak rates	
Intact steam generator	1.0 gpm total
Steam releases	
Faulted steam generator	
0 to 2 hours	9.75E+4 lbs
2 to 8 hours	2.69E+4 lbs
Intact steam generators	
0 to 2 hours	6.53E+5 lbs
2 to 8 hours	1.20E+6 lbs
Average primary coolant flashing factor	0.015
Control room isolation time	10 minutes

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	3658.3 MWt
Radial peaking factor	1.7
Fission product decay period	48 hours
Number of fuel rods damaged	one assembly
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	10 percent
Kr-85	30 percent
I-131	12 percent
Other iodines	10 percent
Fuel pool decontamination factors	
Iodine	100
Noble gases	1
Auxiliary building exhaust system filter efficiency	90 percent
Fuel handling building exhaust system filter bypass	1.0 percent
Duration of accident	2 hours
Control room isolation time	15 seconds

Table 7
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Control Rod Ejection Accident

Parameters	Values
Reactor Power	3658.3MWt
Fuel gap release fraction	10 percent
Fraction of Fuel rods failed	15 percent
Fraction of fuel melt	0.375 percent
Primary coolant activity	60 µCi/gm DEI-131
Secondary coolant activity	0.1 µCi/gm DEI-131
Iodine plate out in containment	50 percent
Containment leak rates	
0 to 24 hours	0.1 percent
1 to 30 days	0.05 percent
Primary coolant mass	2.063E+8 gm
Primary-to-secondary leak rate	1.0 gpm
Iodine partition factor	0.01
Duration of primary-to-secondary leak	1.1 hours
Steam release from secondary	2.5E+6 lbs
Control room isolation time	2.5 minutes

Table 8
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Locked Rotor Accident with Power-Operated Relief Valve Failure

Parameter	Value
Reactor power	3658.3 MWt
Primary coolant iodine activity	60 μCi/gm DEI-131
Secondary coolant iodine activity	0.1 μCi/gm DEI-131
Steam generator tube leak rates	
Faulted steam generator	0.5 gpm
Intact steam generator	0.218 gpm each
Fraction of fuel rods failed	2 percent
Fraction of fission product in gap	10 percent
Iodine partition factors	
steam generators	0.01
PORV release	1
Primary coolant mass	2.063E+8 gm
Duration of PORV release	20 minutes
Duration of steam release	40 hours
Steam release through PORV	
0 to 20 minutes	3.788E+6 gm
After 20 minutes	0
Iodine partition factor used	1
Steam release through steam generators	
0 to 2 hours	2.72E+6 gm
2 to 8 hours	1.40E+6 gm
8 to 40 hours	5/30E+5 gm
Iodine partition factor used	0.01
Control room isolation time	5 minutes

Table 9

Control Room

Parameter	Value
Volume	70,275 ft^3
Emergency ventilation system flow rates	
Filtered makeup air flow	54,000 cfm
Recirculation flow	35,000 cfm
Unfiltered inleakage	100 cfm
Filter efficiencies for intake flow	
Elemental iodine	99 percent
Organic iodine	99 percent
Particulate iodine	99 percent
Filter efficiencies for Recirculation flow	
Elemental iodine	90 percent
Organic iodine	90 percent
Particulate iodine	80 percent
Delay to switch over from normal to emergency operation	15 seconds

Table 10

Meteorological Data

Byron Station

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	5.7E-4

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-8	1.7E-5
8-24	2.4E-6
24-96	1.1E-6
96-720	7.6E-7

Table 11

Meteorological Data

Braidwood Station

Exclusion Area Boundary

Time (hr)	X/Q (sec/m³)
0-2	7.7E-4

Low Population Zone Distance

Time (hr)	X/Q (sec/m³)
0-8	7.1E-5
8-24	1.4E-5
24-96	7.1E-6
96-720	4.1E-6

Table 12

Meteorological Data
Control Room χ/Q (sec/m^3)
for
LOCA - Containment Leak (CL)
LOCA - ECCS Leak
MSLB - Faulted Steam Generator (FSG)
Fuel Handling Accident

Byron Station

Time (hr)	LOCA/CL	LOCA/ECCS and FHA	MSLB/FSG
0-2	6.10E-3	2.28E-3	1.70E-2
2-8	5.30E-3	1.91E-3	1.46E-2
8-24	2.68E-3	8.88E-4	7.24E-3
24-96	2.30E-3	5.97E-4	4.89E-3
96-720	1.53E-3	4.77E-4	3.58E-3

Braidwood Station

Time (hr)	LOCA/CL	LOCA/ECCS and FHA	MSLB/FSG
0-2	6.20E-3	2.48E-3	1.68E-2
2-8	5.37E-3	1.87E-3	1.44E-2
8-24	2.79E-3	8.11E-4	6.53E-3
24-96	1.82E-3	5.04E-4	4.47E-3
96-720	1.32E-3	3.91E-4	2.96E-3

Table 13

Meteorological Data
Control Room χ/Q (sec/m^3)

MSLB/Intact steam generator
SGTR
Locked rotor accident with failed PORV

Time (hr)	Byron	Braidwood
0 to 0.083	8.79E-3	8.71E-3
0.083 to 2	3.98E-3	4.08E-3
2 to 8	3.48E-3	3.43E-3
8 to 24	1.64E-3	1.69E-3
24-96	1.04E-3	9.78E-4
96-720	8.96E-4	6.56E-4

Table 14

Meteorological Data
Control Room χ/Q (sec/m^3)

Rod Ejection Accident - Containment leak (CL)
Rod Ejection Accident -Stem release (SL)

	Byron		Braidwood	
Time (hr)	REA/CL	REA/SL	REA/CL	REA/SL
0 to 0.0417	9.82E-2	8.79E-3	9.53E-2	8.71E-3
0.0417 to 2	6.10E-3	3.98E-3	6.20E-3	4.08E-3
2 to 8	5.30E-3	3.48E-3	5.37E-3	3.43E-3
8 to 24	2.68E-3	1.64E-3	2.79E-3	1.69E-3
24-96	2.00E-3	1.04E-3	1.82E-3	9.78E-4
96-720	1.53E-3	8.96E-4	1.32E-3	6.56E-4