March 15, 2001

MEMORANDUM TO: Anthony Mendiola, Section Chief
Section 2
Project Directorate III
Division of Licensing Project Management

FROM: Frank Akstulwicz, Section Chief ***/RA/***
PWR Reactor Systems Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: BYRON AND BRAIDWOOD STATIONS, UNITS 1 AND 2 - REQUESTS FOR A LICENSE AMENDMENT TO PERMIT UPRATED POWER OPERATIONS (TAC NOS. MA9426, MA9427, MA9428 AND MA9429)

Plant Names: Byron and Braidwood Stations, Units 1 and 2
Utility: Commonwealth Edison Company
TAC Nos: MA9426, MA9427, MA9428, MA9429
Docket Nos: 50-454, 50-455, 50-456, 50-457
Operating License Nos: NPF-37, NPF-66, NPF-72, NPF-77
Project Directorate: PD III-2
Project Manager: George Dick
Review Branch: SRXB/DSSA
Review Status: Complete

By letter dated July 5, 2000, and supplemented by letters dated November 27, 2000, January 31, 2001and February 20, 2001, Commonwealth Edison Company made application to request changes to Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77 and Appendix A, Technical Specifications (TS), for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively. The proposed changes will increase the licensed reactor power from 3411 Mwt to 3586.6 Mwt for these units.

The Reactor Systems Branch (SRXB) has completed its review of the licensee's submittal related to the re-evaluation of the core nuclear and thermal hydraulic design and LOCA and non-LOCA accident analyses. The attached Safety Evaluation covers those areas for which we have primary responsibility. We have concluded that the results of these reevaluations are acceptable to support the proposed power uprates. Gary Hammer of the Mechanical and Civil

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F. Orr, SRXB/DSSA, 415-1815
G. Hammer, MCEB/DE, 415-2791

Engineering Branch assessed the operability of the pressurizer safety valves following either an inadvertent actuation of safety injection event or a main feedwater line break accident and SRXB has included his evaluation in this safety evaluation input. This completes the SRXB effort for TAC Nos. MA9426, MA9427, MA9428, MA9429.

Attachment:
As Stated

cc: S. Bajwa
J. Wermiel
F. Orr
G. Dick
C. Hammer

ACCESSION NUMBER: ML010740316 TEMPLATE #NRR-096
*See previous concurrence page

SRXB:DSSA	EMCB:DE	SRXB:DSSA
CLIANG*	GHAMMER*	FAKSTULEWICZ
2/26 /01	2/26/01	3/15/01

DOCUMENT NAME: G:\BBPURSER2.WPD

ATTACHMENT

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION
CONCERNING LICENSE AMENDMENT TO PERMIT UPRATED POWER OPERATION
COMMONWEALTH EDISON COMPANY
BYRON AND BRAIDWOOD STATIONS UNITS 1 AND 2
DOCKET NUMBERS 50-454, 50-455, 50-456, 50-457

1 INTRODUCTION

By letter dated July 5, 2000, as supplemented by letters dated November 27, 2000, January 31, 2001 and February 20, 2001, Commonwealth Edison Company made application to request changes to Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77 and Appendix A, Technical Specifications (TS), for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively. The licensee has performed a re-evaluation of the core nuclear and thermal hydraulic design and LOCA and non-LOCA accident analyses related to operation at the increased reactor power level of 3586.6 Mwt.

2 EVALUATION

2.1 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, we conclude that the fuel assemblies would not be adversely impacted by the proposed core power uprate.

The reactor coolant systems (RCS) 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 coolant 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 lower flow rate than reviewed by the NRC will not be used. The acceptability of these changes is evaluated in our review of the plant transient and safety analysis results discussed later.

The licensee used the NRC-approved method to evaluate the 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 DNBR values. The current RTDP design limit DNBR values are1.25 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 from1.40, 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 was performed 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 $\Delta k/g/cc$ to 0.54 $\Delta k/g/cc$. These increased values, as well as other nuclear parameter changes (e.g., peaking factors, RCCA worth, reactivity coefficients, shutdown margin and kinetics), are considered in the revised safety analyses described below.

2.2 <u>ACCIDENT ANALYSES EVALUATION</u>

In support of this 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. Except for the analysis of the large break loss of coolant (LBLOCA) to specifically address 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” the uprate program included 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 is scheduled to be submitted in a separate Byron and Braidwood license amendment request in December 2000. **(The words regarding LOCA need be finalized by F. Orr)**

The licensee performed the majority of the uprate analyses and evaluations in accordance with the current Byron Station and Braidwood Station licensing bases methodologies. However, the licensee did perform a number of specific analyses, e.g., the iodine spike factor, LOCA mass and energy release, and feedwater line break calculations using new or improved methods. The staff will discuss the specific analyses in the appropriate safety analysis section of this report. The licensee performed the analyses 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. Although the staff did not formally review and approve this methodology, the staff has used this document as a reference in previous NRC-approved power up-rate submittals by North Anna, Salem, Indian Point 2, Callaway, Vogtle, Turkey Point, and Farley.

2.2.1 <u>LOCA ANALYSIS</u>

(TO BE PROVIDED BY FRANK ORR TO INCLUDE: LBLOCA, SBLOCA, HOTLEG SWITCHOVER, AND POST-LOCA LONG-TERM COOLING)

2.2.2 NON-LOCA ACCIDENT 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 credit 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 departure from nucleate boiling (DNBR) value. The safety analysis limit DNBR was revised from 1.40 to 1.33.

The licensee also revised the over temperature ΔT and overpower ΔT (OTΔT/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 ^oF to 588.0 ^oF. The minimum T_{avg} increased from 569.1 ^oF to 575.0 ^oF. Feedwater temperature at full power conditions increased from 440 ^oF to 446.6 ^oF. The feedwater temperature at hot zero power conditions remains at 100 ^oF. 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 remain 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 determine 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 ^{o}F (i.e., Test #926 with lowest temperature between 635 ^{o}F and 640 ^{o}F and Test #931 with lowest temperatures near 640 ^{o}F) and another performed at a water temperature of approximately 530 ^{o}F (i.e., Test #932). The results of the tests at 635 ^{o}F - 640 ^{o}F show stable valve operation. During the testing at 530 ^{o}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 1 and 2 and Braidwood 1 and 2 is significantly higher (i.e., 590 ^{o}F) than the lowest temperature (i.e., 530 ^{o}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.

2.2.2.1 Excessive Heat Removal Due to Feedwater System Malfunctions

The licensee analysed 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, overpressurization limits for the primary and secondary systems are not challenged for these events. We have 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 find the licensee's analyses acceptable.

2.2.2.2 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. We have 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 find the licensee's analyses acceptable.

2.2.2.3 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 2.2.2.4 and 2.2.2.5 of this report. 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. We have 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 find the licensee's analyses acceptable.

2.2.2.4 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 a 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 and therefore the licensee's analyses would predicted that no fuel failure occur. The licensee's analyses also demonstrates that the calculated peak primary and secondary system pressure do not challenge the allowable peak primary and secondary system pressures. We have 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 find the licensee's analyses acceptable.

2.2.2.5 Steam System 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 demonstrates the 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 would predicted that no fuel failure occur. The licensee's analyses also demonstrates that the calculated peak primary and secondary system pressure do not challenge the allowable peak primary and secondary system pressures. We have 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 find the licensee's analyses acceptable.

2.2.2.6 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 shown that the minimum DNBR remains well 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 shown that the peak primary and secondary systems are maintained below 110 percent of their respective design pressures. We have 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 find the licensee's analysis acceptable.

2.2.2.7 <u>Loss of Non-emergency AC Power to the Plant Auxiliaries</u>

The licensee analyzed the loss of Non-emergency AC power to the plant auxiliaries event at the power uprated conditions using methods that the staff has previouly 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. We have 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 find the licensee's analysis acceptable.

2.2.2.8 <u>Loss of Normal Feedwater</u>

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 DMBR 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 2.2.2.6 of this report. 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. We have 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 find the licensee's analysis acceptable.

2.2.2.9 <u>Feedwater System Pipe Break</u>

The licensee has analyzed the feedwater system pipe break accident at the power uprate conditions. The methodology used for the licensee's analysis is 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 2.2.2 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 lin break analyses since steam blowdown will result in a more excessive cooldown than water blow down 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 2.2.2.6 of this report. The analysis for the bounding loss of load/turbine trip event demonstrate that the peak primary and secondary pressures are maintained below 110 percent of respective design pressures. The results of the licensee's analysis show 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. We have 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 find the licensee's analysis acceptable.

2.2.2.10 Partial Loss of Forced Reactor Coolant Flow

The licensee analyzed A partial loss of reactor coolant flow event (which involve 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 well below 110 percent of their respective design pressures. We have 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 find the licensee's analysis acceptable.

2.2.2.11 Complete Loss of Forced Reactor Coolant Flow

The licensee analyzed two complete loss of forced reactor coolant flow cases at power uprate conditions using methods that the staff has 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. We have 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 find the licensee's analysis acceptable.

2.2.2.12 Single Reactor Coolant Pump Locked Rotor/Shaft Break

The licensee analyzed the single reactor coolant pump lock rotor/shaft break accident at power uprated conditions using methods that the staff has previously approved. The results of the licensee's analysis show that the peak primary and secondary pressures are 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. We have 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 find the licensee's analysis acceptable.

2.2.2.13 <u>Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from a Subcritical Condition</u>

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 ^{o}F). Therefore, no fuel melting or clad damage is predicted as a result of this transient at uprated conditions. We have 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 find the licensee's analysis acceptable.

2.2.2.14 <u>Uncontrolled RCCA Bank Withdrawal at Power</u>

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 and is protected by the high neutron flux and OTΔT trips over the entire range of possible reactivity insertion rates. We have 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 find the licensee's analysis acceptable.

2.2.2.15 <u>RCCA Misoperation</u>

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. We have 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 find the licensee's analysis acceptable.

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

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

2.2.2.17 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant

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. The licensee has stated that the analysis for cold shutdown (Mode 5), hot shutdown (Mode 4), and hot standby (Mode 3) will be supplied in a separate licensing submittal. The event during startup (Mode 2) and power operation (Mode 1) was presented in this submittal. The results of the licensee's analysis indicate that if an unintentional dilution event does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition rate would be slow enough to allow the operator at least 15 minutes to take corrective action before shutdown margin is lost. This is sufficient time to preclude violation of the minimum DNBR limit and the fuel centerline melt limit, and to maintain pressure below 110 percent of design pressure. Therefore, the applicable acceptance criteria for this event continue to be met for the proposed power uprate..We have 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 find the licensee's analysis acceptable

2.2.2.18 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. We have 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 find the licensee's analysis acceptable.

2.2.2.19 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. We have reviewed the results of the licensee's analysis and find it acceptable.

2.2.2.20 Inadvertent Operation of the Emergency Core Cooling System (ECCS) 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 2.2.2 of this report. 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. We have 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 find the licensee's analysis acceptable.

2.2.2.21 Accidental Depressurization of the 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. We have 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 find the licensee's analysis acceptable.

2.2.2.22 Steam Generator Tube Rupture Accident

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 Unit 1 and Unit 2 at Byron and Braidwood Stations. The results of both analyses show that there are sufficient margin to overfill the steam generators prior to operators take control of the auxiliary feedwater flow rate. We have 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 find 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 other Sections of this report.

2.3 TECHNICAL SPECIFICATION CHANGES

2.3.1 - TS 1.1, "Definitions," currently defines RTP as the total reactor core heat transfer rate to the reactor coolant of 3411 Mwt. The value for RTP would be revised to 3586.6 Mwt. The staff has reviewed the analyses documented in the "Power Uprate Licensing Report for Byron Station and Braidwood Station," and concludes that Units 1 and 2 of the Byron and Braidwood

Stations can operate safely with the proposed 5 percent increase in maximum core thermal power. Therefore, the proposed TS change is acceptable.

2.3.2 - TS 2.1.1.1, “Reactor Core Safety Limits,” currently states, “In Mode 1, the Departure from Nucleate Boiling Ratio (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 SL 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.

2.3.3 - TS 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” would be changed to increase the minimum RCS total flow rate from ≥ 371,400 gpm to ≥ 380,900 gpm. Surveillance Requirements (SRs) 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.

3 CONCLUSIONS

Based on our evaluation discussed in Section 2.0 above, the staff has concluded that: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and 2) such activities will be conducted in compliance with the Commission’s regulations, and issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4 REFERENCES

1. Letter from R. Krich, Commonwealth Edison Company to NRC, “Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations,” dated January 5, 2000.

2. Letter from R. Krich, Commonwealth Edison Company to NRC, ”Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations,” dated November 27,2000.

3. Letter from R. Krich, Commonwealth Edison Company to NRC, ”Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations,” dated January 31, 2001.

4. Letter from R. Krich, Commonwealth Edison Company to NRC, “Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Byron and Braidwood Stations,” dated February 20, 2001.