Point Beach Nuclear Plant, Units 1 and 2

Safety Evaluation for Amendment Nos. 207 and 212

Measurement Uncertainty Recapture Power Uprate

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. DPR-27

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application dated April 30, 2002, as supplemented by letters dated June 26, August 29, October 3, October 23, and November 11, 2002, the Nuclear Management Company, LLC (the licensee), requested changes to the Operating Licenses and Technical Specifications (TSs) for the Point Beach Nuclear Plant, Units 1 and 2 (Point Beach). The supplements dated June 26, August 29, October 3, October 23, and November 11, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 11, 2002 (67 FR 57630).

The proposed changes would increase the licensed reactor core power level by 1.4 percent from 1518.5 megawatts thermal (MWt) to 1540 MWt. The proposed increase is considered a measurement uncertainty recapture (MUR) power uprate.

Specifically, the proposed changes would revise:

(1) paragraph 3.A. of Point Beach Facility Operating License Nos. DPR-24 (Unit 1) and DPR-27 (Unit 2) to authorize operation at reactor core power levels up to, but not in excess of, 1540 MWt;

(2) TS 1.1, "Definitions," to state the definition of "RATED THERMAL POWER (RTP)" to reflect the increase from 1518.5 MWt to 1540 MWt; and

(3) TS 5.6.4, "Core Operating Limits Report (COLR)," to add references to Caldon, Inc. (Caldon), Topical Reports ER-80P and ER-160P.

2.0 EVALUATION

2.1 Reactor Systems

2.1.1 Regulatory Evaluation

Point Beach is currently licensed for 1518.5 MWt, and the proposed license amendments would increase the core power level to 1540 MWt.

The proposed amendments are based on the use of the leading edge flowmeter (LEFM) system[1] by Caldon, which improves the accuracy of the feedwater mass flow input to the plant power calorimetric measurement. Caldon Topical Report ER-80P (Reference 1) describes the theory, design, and operating features of the LEFM system. The NRC staff approved Caldon Topical Report ER-80P by a safety evaluation report dated March 8, 1999. The NRC staff reviewed and approved Caldon Topical Report ER-80P for licensees' use in submitting licensing applications for power level increases to 1 percent and for requesting exemptions from certain requirements of 10 CFR Part 50, Appendix K. Caldon supplemented Topical Report ER-80P with Topical Report ER-157P (Reference 2). The NRC staff approved Caldon Topical Report ER-157P by a safety evaluation report dated December 20, 2001.

Since feedwater flow is an essential piece of information in the calculation of the plant operating power output, to gain margin for the uprate, the licensee will reduce the uncertainty in the feedwater flow from 2.0 percent to 0.6 percent. To accomplish this gain, the licensee proposes to install an LEFM system to the feedwater systems for Point Beach. In its loss-of-coolant accident (LOCA) analyses, the licensee used a power level of 2.0 percent. However, recent revisions to 10 CFR Part 50, Appendix K, allow any value for the uncertainty which would account for the uncertainties due to power level instrumentation error.

Chapter 14 of the Point Beach Final Safety Analysis Report (FSAR) contains accident analyses which support the licensee's proposed power level increase to 1540 MWt. The licensee evaluated most of the Chapter 14 accidents assuming a power level 1650 MWt. Even though the current uprate request would only increase power to 1540 MWt, the licensee performed the analyses at 1650 MWt to support a future power uprate request. The NRC staff evaluated the Chapter 14 accidents at their stated power levels.

2.1.2 Technical Evaluation

2.1.2.1 Bounding Non-Loss-of-Coolant Accident Analyses

2.1.2.1.1 Uncontrolled Rod Withdrawal from Subcritical Conditions

An uncontrolled rod cluster control assembly (RCCA) bank withdrawal accident, when the reactor is subcritical, may be caused by a malfunction of the reactor control system or rod control system. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. However, the power range high neutron flux reactor trip (low

[1] LEFM ✓™ 2000FC flow measurement system (LEFM system)

setting) will terminate the accident. The licensee's current analysis assumes the trip takes place at 35 percent of an assumed power level of 1650 MWt.

For this analysis, the licensee used the NRC-approved computer codes TWINKLE, FACTRAN, and THINC. The licensee also used assumptions that ensure conservative results for the startup accident analysis. These assumptions include a conservative Doppler coefficient, initial temperature, moderator reactivity coefficient, initial power, initial effective multiplication factor, reactor trip setpoint, reactivity insertion rate, axial and radial power shapes, and reactor coolant system (RCS) flow. Additionally, the analysis includes a stuck rod with the highest worth reactivity.

The results of the analysis at 1650 MWt indicate that the Standard Review Plan (SRP) acceptance criteria continue to be met (i.e., the minimum departure from nucleate boiling ratio (DNBR) remains above the limit value and the fuel centerline temperatures do not exceed the melting point). Additionally, since the licensee performed this analysis at 1650 MWt using an NRC-approved methodology and conservative assumptions, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.1.2 Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

Similar to the RCCA withdrawal from subcritical conditions, an "uncontrolled RCCA withdrawal at power" accident can be caused by a malfunction of the reactor control or rod control systems. This withdrawal will also uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. For high reactivity insertion rates (the licensee assumes 100 percent mili rho (pcm)/sec), the power range high neutron flux reactor trip terminates the accident. However, for lower reactivity insertion rates from lower power (the licensee assumes 10-percent power and 3 pcm/sec), the overtemperature ΔT ($OT\Delta T$) reactor trip terminates the accident.

For this accident, the licensee assumes a full power level of 1650 MWt and uses the NRC-approved LOFTRAN computer code. The licensee also uses assumptions that ensure conservative results, including initial conditions at 110-percent, 60-percent, and 10-percent power levels, minimum and maximum nominal RCS average temperature, minimum and maximum reactivity feedback, highest worth RCCA stuck, reactor trips for high neutron flux and $OT\Delta T$ at conservative values, and the maximum positive reactivity insertion rate greater than that for the simultaneous withdrawal of two control banks.

The results of the analysis at 1650 MWt indicate that the SRP acceptance criteria continue to be met (i.e., the minimum DNBR remains above the limit value and the fuel centerline temperatures do not exceed the melting point). Additionally, since the licensee performed this analysis at 1650 MWt using an NRC-approved methodology and conservative assumptions, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.1.3 Rod Cluster Control Assembly Drop

RCCA drops occur when the drive mechanism loses power. The drop causes a decrease in power, specifically near the dropped rod, and an increase in the hot channel factors for the remaining rods. As the rod control system tries to restore power to the initial power level, the automatic rod withdrawal will further increase the hot channel factors of the rods. The action of the automatic rod control will cause a power overshoot.

To analyze the automatic rod control case for this accident, the licensee uses a full power level of 1650 MWt, the NRC-approved LOFTRAN and THINC computer codes, and an NRC-approved methodology contained in Licensing Topical Report WCAP-11394-A (Reference 3). The results of the analysis indicate that the SRP acceptance criteria continue to be met because the minimum DNBR for all cases remains above the limit value. Therefore, no fuel failure would occur. Additionally, the peak RCS pressures would remain below the service limit, so RCS pressure boundary integrity would be maintained.

Since the licensee performed this analysis at 1650 MWt using an NRC-approved methodology and approved codes, and since the accident continues to meet the SRP acceptance criteria, the NRC staff finds that the analysis bounds the proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.1.4 Chemical and Volume Control System Malfunction (Boron Dilution)

The licensee evaluated the chemical volume and control system malfunction (boron dilution) event for the uprated power conditions over the spectrum of plant operations, from power operation to refueling. However, changing the analysis power level to 1650 MWt only affected the results of the analysis for power operation. The power level change did not affect the analyses for the other plant conditions.

During power operation at 1650 MWt, with the reactor in automatic control, the power and temperature increases from the boron dilution results in insertion of the RCCAs and a decrease in the shutdown margin. The minimum time to lose the required shutdown margin occurs at the beginning-of-life, and would be greater than 19.0 minutes. However, with the reactor in manual control (assuming no operator action), the power and temperature rises would cause the reactor to trip at the OTΔT high nuclear flux trip setpoint. After the reactor trips, the operators would have 18.2 minutes to prevent a return to criticality.

These times exceed the SRP minimum required operator action time of 15 minutes to terminate dilution during power operation. Since the SRP acceptance criteria continue to be met for this accident at a power level of 1650 MWt, the NRC staff finds that the analysis bounds the proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.1.5 Startup of an Inactive Reactor Coolant Loop

The startup of an inactive loop at the incorrect temperature transient occurs when one reactor coolant pump (RCP) is out of service. With the hot leg temperature of the inactive loop lower than the reactor core inlet temperature, this startup results in the injection of cold water into the core. This injection causes a reactivity insertion and subsequent power increase.

However, the Point Beach TSs do not allow for critical operation with an inactive reactor coolant loop. Therefore, the NRC staff finds that this accident does not need to be reevaluated for the uprated conditions.

2.1.2.1.6 Reduction in Feedwater Enthalpy Incident

A reduction in feedwater enthalpy incident occurs when relatively cool feedwater is supplied to the steam generators (SGs). This action causes excess heat removal by the secondary side, which increases core power above full power. This accident could occur if the feedwater bypass valve is accidentally opened. This valve then diverts flow around the low pressure feedwater heaters, and in turn, delivers feedwater with lower enthalpy to the SGs. However, the protection afforded by the overpower-overtemperature protection of the RCS (nuclear overpower and ΔT trips) prevents a power increase that could lead to a violation of the DNBR limits.

The licensee evaluated this accident at a power level of 1518.5 MWt, and determined that the feedwater enthalpy reduction results in a transient that is very similar to, but bounded by that of an excessive load increase transient. Because the excessive load increase transient bounds this event, the licensee determined that it does not need to be reanalyzed for the uprated conditions. The NRC staff agrees with the licensee's assessment that the reduction in feedwater enthalpy incident does not need to be reanalyzed for the uprated conditions.

2.1.2.1.7 Excessive Load Increase Incident

An excessive load increase incident occurs when a rapid increase in steam flow causes a power mismatch between the reactor core power and the SG load demand. The RCS accommodates a 10-percent step load increase or a 5-percent per minute ramp load increase between 15-percent and 100-percent power. However, loading rates exceeding these values may result in a reactor trip initiated by the reactor protection system (RPS). The reactor trip, caused by either the overpower trip or the overpower-overtemperature (opΔT) trips, keeps the minimum DNBR above the limiting value.

Because the excessive load increase incident does not lead to a serious challenge to the acceptance criteria, the licensee did not reanalyze this transient for a core power of 1650 MWt. However, the licensee evaluated the state points for the load increase for a core power of 1650 MWt, using 422V+ fuel. The licensee's evaluation confirmed that the core thermal limits are not challenged by this event, and the minimum DNBR would remain above the safety analysis limit value.

Because the DNBR values for this accident remain above their limit values, the transient would continue to meet the SRP acceptance criteria at a power level of 1650 MWt. Therefore, the NRC staff finds the licensee's evaluation acceptable for the proposed power level of 1540 MWt.

2.1.2.1.8 Loss of Reactor Coolant Flow and Locked Rotor (Core Response)

A mechanical or electrical failure in one or more RCPs or a fault in the power supply to these pumps may cause a partial or complete loss of forced coolant flow. If the reactor is powered at the time of the incident, the loss of coolant flow causes a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB).

The licensee performed an evaluation of the accident for a power level of 1650 MWt. This meets the SRP acceptance criteria such that the system maintains pressure in the reactor coolant and main steam systems below 110 percent of design values and DNB does not occur. Consequently, the current design basis remains valid and the NRC staff finds the licensee's assessment acceptable.

An RCP locked rotor accident, on the other hand, results from the instantaneous seizure of an RCP rotor. The flow through the affected reactor coolant loop rapidly reduces, and the reactor trips on a low reactor coolant flow signal. The sudden decrease in core coolant flow while the reactor is powered results in a decreased core heat transfer, which may result in fuel damage.

The licensee performed an evaluation of the accident for a power level of 1650 MWt, assuming one locked rotor and one RCP coasting down using conservative operating conditions. The licensee did not assume a loss of offsite power. The licensee showed that it met the SRP acceptance criteria because the maximum RCS pressure only reaches 2873 psia, which is lower than their faulted condition stress limits of 3120 psia. Also, their peak cladding temperatures reach 2124 °F, and the zirconium-water reaction is small (1.1 percent by weight at the core hot spot), therefore, the licensee concludes that the core would remain in place and intact with no consequential loss of core cooling capability. Although DNB would occur for this accident, because the core would remain in a coolable geometry, the current design basis would remain valid. Therefore, the NRC staff finds the licensee's assessment acceptable.

2.1.2.1.9 Loss of External Electrical Load (Overpressure)

A loss of external electrical load event occurs when an electrical disturbance causes the loss of a significant portion of the generator load. The Point Beach units are designed to accept a 50-percent loss of electrical load while operating at full power, without actuating a reactor trip. However, if a full loss of load occurs, the RPS automatically initiates a reactor trip. The most likely source of a complete loss of load on the nuclear steam supply system (NSSS), on the other hand, is a turbine-generator trip. The licensee evaluated this complete loss of load from full power without a direct reactor trip to ensure that the pressure relieving devices work, and to ensure that no core damage occurs.

To determine the minimum DNBR and maximum peak RCS pressures achieved, the licensee used the NRC-approved LOFTRAN computer code. As inputs into the analyses, the licensee used assumptions that ensure conservative results. For the DNB calculations, the licensee assumed a power level of 1650 MWt, with uncertainties of 2 percent for power, 50 psi for

RCS pressure, and 6 °F for RCS temperature. However, for the overpressure calculations, the licensee assumed a power level of 102 percent of 1650 MWt.

The results of the analyses for the overpressure case and the DNB case indicate that the SRP acceptance criteria continue to be met (i.e., the minimum DNBR remains above the limit value and the peak RCS pressure is within 110 percent of design limits). Additionally, since the licensee performed these analyses using NRC-approved codes and methodologies for a power level of 1650 MWt, the NRC staff finds that the analyses bound the proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analyses are acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.1.10 Loss of Normal Feedwater

A loss of normal feedwater event reduces the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped, or if an alternate supply of feedwater were not supplied to the plant, core damage could occur. Currently, the Point Beach loss of normal feedwater analysis uses the NRC-approved LOFTRAN computer code to model a power level of 102 percent of 1518.5 MWt (1549 MWt). This power level bounds the requested uprate level of 1540 MWt with a 0.6-percent uncertainty. Since the current analysis bounds the power uprate, the NRC staff finds it acceptable.

2.1.2.1.11 Loss of All AC Power to the Station Auxiliaries

The loss of all AC power to the station auxiliaries transient results in a loss of all power to auxiliary systems (including the RCPs, condensate pumps, etc). Upon the loss of power, core cooling and removal of residual heat is accomplished by natural circulation in the reactor coolant loops, aided by auxiliary feedwater, SG PORVs, and safety valves on the secondary side.

For this analysis, the licensee used the NRC-approved LOFTRAN computer code with input assumptions that ensure conservative results. The acceptance criteria for this transient include preventing the minimum DNBR from going below the limit value, preventing the RCS pressure from going above 110 percent of the design value, and preventing pressurizer overfill.

For the DNBR and RCS pressure criteria, the loss of external electrical load transient bounds this event. Therefore, the licensee specifically analyzed the loss of AC power to the station auxiliaries transient for pressurizer overfill. The results of the licensee's analysis demonstrate that at a core power level of 1650 MWt, the Point Beach pressurizers do not overfill.

The analysis at a core power of 1650 MWt indicates that the acceptance criteria continue to be met (i.e., the minimum DNBR, RCS pressure, and pressurizer overfill). Additionally, since the licensee performed the analysis at 102 percent of 1650 MWt core power using an NRC-approved methodology and conservative assumptions, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.2 Bounding Non-Loss-of-Coolant Accidents with Potential Radiological Consequences

2.1.2.2.1 Rupture of a Steam Pipe (Core Response)

The rupture of a steam pipe accident models an uncontrolled steam release from an SG, which includes steam pipe breaks and valve malfunctions. The most limiting steam pipe accidents occur when the reactor is in a hot shutdown condition. With the RCS in this condition, the steam release will cool the RCS. Since the RCS has a negative moderator temperature coefficient, this cooling may cause the core to become critical and return to power, possibly causing fuel damage. The safety injection (SI) system eventually terminates this accident by supplying boric acid to shut down the core.

Because the most limiting case of this accident occurs at no load conditions, and because the SI system terminates the accident independent of power level, the core response portion of the steam pipe rupture accident remains independent of power level. Since the core response of this accident is not influenced by power level, the NRC staff finds the core response acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.2.2 Rupture of Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly Ejection (Core Response)

A control rod drive mechanism (CRDM) pressure housing rupture may result in the ejection of an RCCA and drive shaft to their fully withdrawn position. The consequences of this failure include a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage.

For this analysis, the licensee used the NRC-approved codes TWINKLE, FACTRAN, and THINC. In addition, the licensee analyzed the RCCA ejection at the beginning and end of the core life for both hot full power and hot zero power conditions in order to bound the fuel cycle and expected operating conditions. The licensee also used assumptions that ensure conservative results for the accident analysis.

For the hot full power condition, the licensee reevaluated its accident at a power level of 102 percent of 1650 MWt. The licensee also reevaluated the accident at zero-percent power for the hot zero power case. The licensee's evaluation showed that the accident acceptance criteria continue to be met because the limiting fuel pellet enthalpies remain below 200 cal/g. Also, the total rods in DNB continue to be less than 15 percent, and the peak pressures remain below the faulted condition stress limits. Since the analysis uses conservative assumptions and continues to meet the SRP acceptance criteria at an operating power level of 1650 MWt, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the1.4-percent MUR power uprate.

2.1.2.3 Bounding Loss-of-Coolant Accident Related Analyses

2.1.2.3.1 Small-Break Loss-of-Coolant Accident

The licensee performed the small-break LOCA (SBLOCA) analysis assuming 102 percent of a 1650 MWt power level (1683 MWt). For this event, the licensee used the Westinghouse

NOTRUMP SBLOCA analysis methodology described in Licensing Topical Report WCAP-10054-P-A (Reference 4). The NRC approved the licensee's use of the methodologies in WCAP-10054-P-A by License Amendment Nos. 193 and 198 for Point Beach, Units 1 and 2, respectively (Reference 6). License Amendment Nos. 193 and 198 approved the licensee's application dated June 22, 1999, which was supplemented by letter dated December 17, 1999. In its supplemental letter dated December 17, 1999, the licensee stated that it had ongoing processes to assure that the analysis input values for peak-cladding-temperature-sensitive parameters conservatively bound the as-operated plant values for those parameters.

In its review of the proposed 1.4-percent MUR power uprate, the NRC staff also considered the continued applicability of the methodologies' constituent models at the uprated power conditions. The licensee's processes for determining input values for the methodology, stated above, resolved this concern by assuring continued conservatism of the SBLOCA methodology at a power level of 1650 MWt.

In its June 22, 1999, application proposing the use of Vantage 422+ fuel, the licensee also provided the SBLOCA analysis results at 102 percent of 1650 MWt. The analysis results indicated that peak cladding temperatures reach 1157 °F, maximum local cladding oxidation values reach 0.03 percent, and maximum core-wide oxidation values reach 0.01 percent. These values are within the emergency core cooling system (ECCS) acceptance criteria given in 10 CFR 50.46(b)(1)-(3). From these results, the licensee concluded that the core remains amenable to cooling, as required by 10 CFR 50.46(b)(4).

Since the results of the analysis at 102 percent of 1650 MWt indicate that the 10 CFR 50.46 ECCS acceptance criteria continue to be met for this accident, and since the licensee performed the analysis using an NRC-approved methodology, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

2.1.2.3.2 Large-Break Loss-of-Coolant Accident

The licensee performed the large-break (LBLOCA) analysis assuming 102 percent of a 1650 MWt power level (1683 MWt). For this event, the licensee used the Westinghouse two-loop, upper plenum injection (UPI) version of the NRC-approved Westinghouse best-estimate LBLOCA analysis methodology described in Licensing Topical Report WCAP-14449-P-A (Reference 5). The NRC approved the licensee's use of the methodology in WCAP-14449-P-A by License Amendment Nos. 193 and 198 for Point Beach, Units 1 and 2, respectively (Reference 6). As noted above, the licensee stated in its December 17, 1999, letter that it had ongoing processes to assure that the analysis input values for peak-cladding-temperature-sensitive parameters conservatively bound the as-operated plant values for those parameters.

In its review of the proposed 1.4-percent MUR power uprate, the NRC staff also considered the continued applicability of the methodologies' constituent models at the uprated power conditions. The licensee's provision of the ongoing processes stated above resolved this concern by assuring continued conservatism of the LBLOCA methodology at a power level of 1650 MWt.

In its June 22, 1999, application, the licensee also provided the LBLOCA analysis results at 102 percent of 1650 MWt. The analysis results indicated that peak cladding temperatures reach 2128 °F, maximum local cladding oxidation values reach 8.52 percent, and the maximum hydrogen generation values reach 0.0081 times the maximum theoretical amount. These values are within the criteria given in 10 CFR 50.46(b)(1)-(3). From these results, the licensee concluded that the core remains amenable to cooling, as required by 10 CFR 50.46(b)(4).

Since the results of the analysis at 102 percent of 1650 MWt indicate that the 10 CFR 50.46 ECCS acceptance criteria continue to be met for this accident, and since the licensee performed the analysis using an NRC-approved methodology, the NRC staff finds that the analysis bounds the licensee's proposed power level of 1540 MWt. Therefore, the NRC staff concludes that the analysis is acceptable for the licensee's proposed 1.4-percent MUR power uprate.

Long-Term Core Cooling

The regulation at 10 CFR 50.46(b)(5) establishes the long-term cooling requirement following a LOCA. One issue with long-term cooling is ensuring that boric acid (H_3BO_3) accumulation will not prevent core cooling. Because of boron precipitation, the NRC found that changes to operating procedures at some plants were needed to ensure adequate hot leg switch-over times.

The licensee states that Point Beach is a UPI plant, where the low head ECCS pumps (residual heat removal (RHR) pumps) deliver flow to the core deluge nozzles directly into the upper plenum. Because of this design, the licensee states that the hot leg switchover procedure that is applied at some Westinghouse plants does not apply for Point Beach. The licensee maintains that as long as the RCS pressure is below the RHR shutoff head, the RHR pumps would continuously inject ECCS fluid directly above the core. This injection, in turn, would cause flushing and mixing of the upper plenum and upper core. Therefore, the licensee concludes that boron precipitation during the long-term cooling phase of an LBLOCA should not occur. Since Point Beach is a UPI plant, the NRC staff concludes that boron precipitation is not an issue during the design-basis LBLOCA. Therefore, the NRC staff finds the licensee's assessment acceptable for the 1.4-percent MUR power uprate.

2.1.2.4 Other Bounding Licensing-Basis Analyses

2.1.2.4.1 Natural Circulation Cooldown

The goal of coping with a natural circulation cooldown event is to prevent voiding in the upper head of the RCS pressure vessel. Upon evaluation of this event at a core power level of 1518.5 MWt for Point Beach, the licensee demonstrated that it had adequate condensate supplies to cool down the RCS if the CRDM fans are available. The licensee also demonstrated that means are available to obtain condensate supplies if necessary. The licensee performed this calculation with an 8-percent margin on their ANSI/ANS-5.1-1979 decay heat values. Since only 4 percent of this margin is required to account for the 2σ uncertainty associated with the decay heat, the licensee has an extra 4-percent margin available. Following the "Simplified Method for Determining Decay Heat Power and Uncertainty" of ANSI/ANS5.1-1979, Section 3.6, the licensee applied the remaining

4-percent margin to account for the MUR power uprate. This method shows that decay heat varies linearly with reactor thermal power. Therefore, the licensee's extra 4-percent margin for decay heat equates to a 4-percent margin for thermal power. Since the licensee's calculation has an extra 4-percent margin for power above 1518.5 MWt, it bounds the requested MUR power uprate of 1540 MWt with an 0.6 percent uncertainty. Therefore, the NRC staff finds this analysis acceptable for the proposed 1.4-percent MUR power uprate.

2.1.2.4.2 Anticipated Transient Without Scram

The licensee analyzed the anticipated transient without scram (ATWS) event for the 1.4-percent MUR power uprate as required by 10 CFR 50.62, "Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants" (ATWS Rule). The ATWS Rule for Westinghouse plants requires that the plant must implement a system diverse from the reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS. The system is called the ATWS mitigation system action circuitry (AMSAC) system. Point Beach meets the requirements of 10 CFR 50.62(c) by having an AMSAC system with an appropriate setpoint of 30 seconds before the auxiliary feedwater system actuates and the turbine trips.

Additionally, to demonstrate that the proposed MUR power uprate would not result in a transient peak RCS vessel pressure above the ASME stress level C limit of 3200 psig, the licensee relied upon the Westinghouse generic ATWS analysis (Reference 7). The licensee determined that the loss of external load and the complete loss of normal feedwater events are the most limiting ATWS events for RCS pressure.

The generic loss-of-external-load ATWS analysis for a two-loop pressurized-water reactor (PWR) with Model 44 SGs and a core power of 1520 MWt applies best to Point Beach. Both Point Beach units are two-loop PWRs. Currently, Unit 1 has Westinghouse Model 44F SGs and Unit 2 has Westinghouse Delta 47 SGs. The licensee determined that the performance characteristics of these SGs closely resemble those of the Model 44 SGs. The licensee also relied upon sensitivity analyses (including a 2-percent increase in reactor power) performed for a 4-loop plant with Model 51 SGs to account for differences between the Point Beach units and their reference plant (two-loop, Model 44 SGs). The pressure results for the 4-loop plant bound those of the two-loop plants with Model 44 SGs. The licensee's evaluation shows that the peak pressure would not exceed 2903 psia for the loss-of-external-load ATWS.

The generic loss of normal feedwater ATWS analysis for a 3-loop PWR with Model 44 SGs and a core power of 1520 MWt applies best to Point Beach. For this event, the licensee determined that the 3-loop analysis bounds the two-loop event. Using the same methodology as for the loss-of-external-load ATWS, the licensee determined that the peak pressure would not exceed 2789 psia for the loss of normal feedwater ATWS. Both ATWS events analyzed give peak pressures below the limit of 3200 psig, therefore, the NRC staff finds the licensee's assessment acceptable to support the proposed 1.4-percent MUR power uprate.

2.1.2.4.3 Station Blackout

In their coping analysis for a station blackout (SBO) event, the licensee performed its calculations assuming a power level of 1518.5 MWt with a 4.5-percent uncertainty, which equates to 1587 MWt. Since this analysis continues to bound the proposed power level of

1540 MWt with a 0.6-percent uncertainty (1549 MWt), the NRC staff finds the analysis acceptable for the requested MUR power uprate.

2.1.2.5 Nuclear Steam Supply System Design Parameters

The NSSS design parameters provide the RCS and secondary system conditions for use in the NSSS analyses and evaluations. The licensee presented parameters for the power levels of 1518.5 MWt, 1540 MWt, and 1650 MWt. Even though all the accident analyses and component analyses used either the 1518.5 MWt or the 1650 MWt parameters, the licensee provided the 1540 MWt parameters for comparison purposes. The key parameters included reactor power, RCS pressure, T_{avg} range, thermal design flow, steam pressure, steam temperature, and steam flow rate. The differences between the parameters at 1518.5 MWt and 1650 MWt included an increased core power level, increased T_{avg}, lower thermal design flow, higher SG tube plugging, lower steam pressure, lower steam temperature, and a higher steam flow rate. The NRC staff evaluated these changes to the plant conditions and found them to adequately represent the plant behavior at the specified power levels; therefore, the NRC staff finds the NSSS design parameters to be acceptable.

2.1.2.6 Reactor Vessel Integrity

This section discusses neutron irradiation, heatup and cooldown curves, low temperature overpressurization (LTOP), and pressurized thermal shock (PTS).

The licensee requested that the current pressure-temperature (P-T) limit curves and the low temperature overpressure (LTOP) protection limit setpoints be approved for use until the next P-T update. In its August 29, 2002, supplemental letter, the licensee restated its commitment to revise the P-T and LTOP limits before October 1, 2003. This analysis must be included in the Pressure and Temperature Limits Report (PTLR) for each reactor vessel fluence period in accordance with TS 5.6.5. Consistent with the methodology approved in TS 5.6.5, and the licensee's letter dated August 29, 2002, the licensee will revise the P-T and LTOP limits before October 1, 2003. In accordance with TS 5.6.5, the licensee will provide the PTLR to the NRC upon its issuance. In addition, the licensee provided information to demonstrate the validity of the current values of the maximum nil ductility reference temperature in the limiting beltline component, termed reference-temperature pressurized thermal shock (RT_{PTS}) at the uprated power.

Both Point Beach units have common P-T curves located in the Point Beach PTLR. The PTLR lists NRC-accepted methodologies to be used in the implementation of P-T curve changes. In addition, the licensee stated that the methodology being used for fluence calculations, which is listed in the PTLR, adheres to the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Therefore, the NRC staff concludes that the methodology is acceptable.

The current P-T curves were designed for 2.25×10^{19} n/cm^2 (for Unit 1) and 2.61×10^{19} n/cm^2 (for Unit 2). These values correspond to 25.59 effective full power years (EFPYs) of operation for Unit 1 and 30.51 EFPYs for Unit 2. By November 2002, Unit 1 will have accumulated 24.42 EFPYs and Unit 2 will have accumulated 24.02 EFPYs. The licensee proposes to update the P-T curves in October 2003. By October 2003, Unit 1 will have accumulated 25.34 EFPYs and Unit 2 will have accumulated 24.94 EFPYs. Both values are within the approved range of

operation, therefore, the NRC staff finds the existing P-T limits acceptable for use until October 2003. After this date, the requirements of TS 5.6.5 will govern the calculation of P-T limit curves and LTOP setpoints.

The LTOP limits are also located in the PTLR, and the methods, like those of the P-T limits, have the same applicability limits. The LTOP would be revised along with the P-T limits. The current LTOP limits are acceptable until October 2003.

Finally, the licensee calculated the PTS limiting RT_{PTS} values using a power level of 1678 MWt (about 10 percent greater than the current power level). The RT_{PTS} values for both units are within the limits of 10 CFR 50.61. Additionally, the fluence values to 32 EFPY have been calculated using NRC-approved methods and the values are conservative with respect to the proposed 1.4-percent MUR power uprate. Therefore, the values are acceptable and do not need updating (i.e., like the P-T and LTOP values).

In summary, the NRC staff reviewed the information submitted by the licensee to determine the applicability of the P-T and LTOP and to assert the validity of RT_{PTS}. The NRC staff determined that the fluence calculations were performed in accordance with a previously-approved methodology, and therefore, the calculated values are acceptable. Based on this conclusion, the NRC staff also finds that the P-T limits and the LTOP limits are valid until October 2003. In addition, RT_{PTS} meets the screening criteria of 10 CFR 50.61 for both units until the end of the current license.

2.1.2.7 Safety-Related Cooling Water Systems

2.1.2.7.1 Safety Injection System

The licensee evaluated the SI system up to a power level of 1678 MWt and determined that the system remains unchanged because of the 1.4-percent MUR power uprate. Since the system would remain acceptable up to a power level of 1678 MWt, the NRC staff finds that it would remain acceptable for the uprated power level of 1540 MWt with a 0.6-percent uncertainty.

2.1.2.7.2 Residual Heat Removal System

The licensee evaluated the RHR system for both 10 CFR Part 50, Appendix R, and normal cooldown requirements for a power condition of 1549 MWt. The licensee found that the system would remain adequate and the system requirements would continue to be met at this condition. Since the RHR system would remain acceptable at 1549 MWt, the NRC staff finds that it would remain acceptable for the uprated power level of 1540 MWt with a 0.6 percent uncertainty.

2.1.2.8 Reactor Protection System Settings

Upon examination of the RPS settings of TS 3.3.1, the licensee determined that several functions of the RPS could be affected by a power uprate to 1540 MWt. These functions include the power range neutron flux high and low, intermediate range neutron flux, overtemperature ΔT, and overpower ΔT ($OP\Delta T$) trip setpoints.

The power range neutron flux high setpoint trips the reactor at 108-percent RTP, and the low setpoint trips the reactor at 25 percent RTP. Similarly, the intermediate range neutron flux trip setpoint activates at a power less than or equal to 40 percent RTP. Because these three setpoints are based upon RTP, the licensee determined that the setpoints would need to be scaled for the uprated power of 1540 MWt. Since the setpoints are based upon rated power, the NRC staff agrees with the licensee's assessment that the setpoints need to be scaled for the power uprate.

The OTΔT and OPΔT functions both have an input from the ΔT at rated power (ΔT_o). Since this value changes for the new power, the licensee determined that it needs to change its procedure for calibrating ΔT_o. The licensee also determined that it needs to modify the COLR to incorporate this change. Since the ΔT inputs for the setpoints change based upon power, and since the Point Beach COLR includes input from the OTΔT and OPΔT trips, the NRC staff agrees with the licensee's assessment.

In addition to the above RPS trip functions, the steam flow/feedwater flow mismatch reactor trip input parameters also change because of a power level change. However, since both the steam flow and feedwater flow change equivalently, their difference would remain unchanged. Since this difference is the input to the trip function, the licensee determined that the trip function would not need to be modified to support the MUR power uprate. Because the values for the trip function remain unaffected by the uprated power, the NRC staff agrees with the licensee's determination.

2.1.2.9 Steam Generator Water Level Trip Setpoints

Recently, Westinghouse identified issues with SG water level setpoint uncertainties. One problem deals with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG water level measurements. These uncertainties affect the low-low level trip setpoint.

The licensee evaluated the effects of a pressure drop across the mid-deck plate on the SG low-low level trip setpoint. However, because of the large flow areas present in the mid-deck plate for the Point Beach SGs, the licensee concluded that the pressure drop is essentially zero. Therefore, the licensee determined that no corrections are required for the low-low level trip setpoint. Because the pressure drop across the plate is essentially zero, the NRC staff agrees that the low-low level trip setpoint does not need to be adjusted because of the mid-deck plate effects.

Additionally, another issue includes uncertainties created because the effects of the void content of the two-phase mixture above the mid-deck plate were not reflected in the calculation and may affect the high-high level trip setpoint. The licensee showed that as the SG water level reaches and passes the high-high level trip (including trip level uncertainties), the water level is still below the elevation of the mid-deck plate. Therefore, the licensee concludes that the void fraction above the mid-deck plate does not impact the acceptability of the high-high SG trip. Since the high-high SG trip activates when the water level is below the mid-deck plate, the NRC staff agrees with the licensee's assessment that the void fraction above the mid-deck plate does not impact the high-high trip setpoint.

The final issue pertains to the initial conditions assumed in the safety analyses related to the SG water level. These analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which may result in significant increases in the control system uncertainties.

As noted above, the Point Beach units are not affected by mid-deck plate pressure differentials because they have an essentially zero pressure drop across the mid-deck plate. However, they were affected by the velocity head effects on the indicated SG water level. The licensee considered these effects during its Unit 1 Level Tap Relocation Program and Unit 2 Replacement Steam Generator Program. Under these programs, the licensee used the velocity head term in its calculation for total loop error. Since the licensee has a zero pressure drop mid-deck plate, and since it included velocity head effects in its total error calculations for the SG level measurements, the NRC staff finds that the licensee adequately addressed this issue.

Because the above issues do not impact the current Point Beach SG level trip setpoints, the NRC staff finds that the setpoints the licensee used in its accident analyses remain acceptable.

2.1.2.10 Emergency Settings

Upon examination of the engineered safety feature actuation system instrumentation functions of TS Table 3.3.2-1, the licensee determined that two functions could be affected by a power uprate to 1540 MWt: (1) Function 4.d, “Steam Line Isolation” on “High Steam Flow” and (2) Function 4.e, “Steam Line Isolation” on “High High Steam Flow.”

The Point Beach high steam flow setpoint (4.d) currently allows a value up to 0.66 E 06 lb/hr at 1005 psig. The proposed 1.4-percent uprate would increase this value up to 0.669 E 06 lb/hr. However, the licensee proposed to keep the allowable value at 0.66 E 06 because it would have the effect of tripping earlier while maintaining an adequate margin from spurious plant trips. Since tripping earlier is a conservative change, and since adequate margin exists from spurious plant trips, the NRC staff finds the high steam flow (4.d) allowable value acceptable.

The 1.4-percent MUR power uprate similarly affects the high steam flow allowable values (4.e). Currently, the value is based on 120 percent of full steam flow at full steam pressure. This value equates to 4 E 06 lb/hr at 806 psig. The 1.4-percent MUR power uprate would increase this value to 4.06 E 06 lb/hr. Similar to the high steam flow allowable value, tripping earlier would be a conservative change, and adequate margin exists from spuriously tripping the plant. Therefore, the licensee proposed to keep the value at 4 E 06 lb/hr. The NRC staff finds this value acceptable for the high steam flow setpoint (4.e).

2.1.3 Reactor Systems Conclusion

The NRC staff has reviewed the licensee’s analyses to support operation of Point Beach at a maximum core power level of 1540 MWt. Based on this review, the NRC staff finds that: (1) the supporting safety analyses were performed using NRC-approved methods, (2) the input parameters of the analyses adequately represent the plant conditions at the uprated power level, and (3) the analytical results meet the applicable acceptance criteria. Therefore, the NRC staff concludes that the supporting analyses are acceptable for the 1.4-percent MUR power uprate.

2.2 Civil and Engineering Mechanics

2.2.1 Regulatory Evaluation

The civil and engineering mechanics evaluation covers the structural integrity of pressure-retaining components, their supports, and core support structures, which are designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code). The following specific editions and addenda of Section III of the B&PV Code were used.

- 1965 edition for Unit 1 and 1968 edition with addenda through winter 1968 for Unit 2
- Appendix F, 1974 edition
- Subsection NG, 1989 edition
- 1968 edition with addenda through winter 1969
- 1965 edition through the summer 1966 addendum
- 1968 edition through the summer 1968 addendum
- USAS B31.1, Power Piping Code
- 1986 edition

Additional information is also provided by the plant-specific evaluations of Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

2.2.2 Technical Evaluation

The NRC staff reviewed the licensee's application for the 1.4-percent MUR power uprate as it relates to the effects of the MUR power uprate on the structural and pressure boundary integrity of the NSSS and balance-of-plant (BOP) systems. Affected components in these systems included piping, in-line equipment and pipe supports, the reactor pressure vessel (RPV), core support structures (CSS), reactor vessel internals (RVI), SGs, CRDMs, RCPs, and the pressurizer.

2.2.2.1 Reactor Vessel

The proposed MUR power uprate would increase the core power by approximately 1.4 percent above the currently licensed level of 1518.5 MWt. The licensee reported that the power increase would result in changing the design parameters given in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application. Table 3.4.1-1 provides a comparison of the current design parameters, the revised design parameters at the proposed uprated power level of 1540 MWt, and the design parameters at the core power level of 1650 MWt that were used in the bounding power uprate analysis for the proposed MUR power uprate.

The licensee evaluated the reactor vessel for the effects of the revised design conditions provided in Table 3.4.1-1. The evaluation was performed for the limiting vessel locations with regard to stresses and fatigue cumulative usage factors (CUFs) in each of the regions, as identified in the reactor vessel stress reports for the core power uprated conditions. 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, SI nozzles, external support brackets, the bottom head-to-shell juncture, core support guides, and the instrumentation tubes. In its August 29, 2002, supplemental letter responding to an NRC staff request for additional information (RAI), the licensee indicated that the evaluation of the reactor vessel was performed in accordance with the code of record for Point Beach, which is the ASME B&PV Code, Section III, 1965 edition for Unit 1, and 1968 edition with addenda through winter 1968 for Unit 2. However, the evaluation for the faulted conditions was performed based on standards of Appendix F of the ASME B&PV Code, Section III, 1974 edition, because Appendix F was not available in the 1965 and 1968 editions. The NRC staff finds this acceptable in accordance with 10 CFR 50.55a.

In Attachment 5 to its August 29, 2002, supplemental letter, the licensee provided the calculated maximum stresses and the maximum CUFs for the reactor vessel critical locations. The results indicate that the maximum primary plus secondary stresses are within the code-allowable limits, and the CUFs remain below the code-allowable limit of 1.0. Therefore, the NRC staff agrees with the licensee's conclusion that the current design of the reactor vessel continues to be in compliance with licensing-basis codes and standards for the proposed power uprate condition.

2.2.2.2 Reactor Core Support Structures and Vessel Internals

In its August 29, 2002, supplemental letter, the licensee provided additional information requested by the NRC 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, lower support columns, core barrel, baffle plates, baffle/barrel region bolts, guide tubes and support pins, and the upper core plate. The licensee indicated that the reactor internal components were not licensed to the ASME B&PV Code. However, the design of the Point Beach reactor internals was evaluated in accordance with standards of Subsection NG of the 1989 edition of the ASME B&PV Code, Section III. This is acceptable to the NRC staff in accordance with 10 CFR 50.55a.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application for a core power of 1650 MWt, which is bounding for the proposed power level of 1540 MWt. Table 2-1 of Attachment 5 to the licensee's August 29, 2002, supplemental letter provides the margins of safety and CUFs for the limiting reactor internals. The calculated margins of safety as shown in the table are positive, indicating that the calculated stresses are less than the code-allowable stress limits. The calculated CUFs are less than the ASME code-allowable limit of 1.0. The remaining reactor internal components are less limiting. In addition, the licensee evaluated flow induced vibration, which was found to remain within the allowable limits for the power uprate condition. Based on the above evaluation, the NRC staff agrees with the licensee's conclusion that the reactor internal components at Point Beach will be structurally adequate for the proposed MUR power uprate.

2.2.2.3 Control Rod Drive Mechanisms

The pressure boundary portion of the CRDMs are those exposed to the vessel/core inlet fluid. Both Point Beach units have L-106A CRDMs, full-length mechanisms manufactured by Westinghouse. The licensee evaluated the adequacy of the CRDMs by reviewing the original E-Specification and the generic evaluation for L-106A CRDMs to compare the design-basis input parameters against the revised design conditions in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application for the power level of up to 1650 MWt. The licensee indicated in Attachment 5 to its August 29, 2002, supplemental letter that the key input parameters (e.g., the hot leg maximum temperature, the maximum pressure fluctuation, and the maximum temperature fluctuation) for the uprated power condition are bounded by the design-basis analysis. The power uprate evaluation was performed using Section III of the ASME B&PV Code, 1968 edition with addenda through winter 1969. Table 3-1 of Attachment 5 to the August 29, 2002, supplemental letter provides the calculated CUFs for the critical CRDM locations at the proposed power uprate conditions, which are less than the code-allowable limit of unity.

As a result of its review which is described above, the NRC 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 proposed 1.4-percent MUR power uprate.

2.2.2.4 Steam Generators

The licensee reviewed the existing structural and fatigue analyses of the SGs at Point Beach and compared the power uprate conditions with the design parameters of the analysis of record for the Model 44F SGs at Unit 1, and of the Model Delta-47 SGs at Unit 2. The comparison of key parameters for power levels of up to 1650 MWt are shown in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application. The licensee evaluated SG components in accordance with the standards of the ASME Code, Section III, 1965 edition through the summer 1966 addendum, which is the code of record for Point Beach.

In its August 29, 2002, supplemental letter, the licensee provided the calculated CUFs for the critical SG components for the uprated power conditions. The licensee indicated that finite element analysis was performed for the evaluation of SGs at Unit 2 during the Replacement Steam Generator Program. For the evaluation of Unit 1 SGs, the licensee incorporated the key input parameters to develop scaling factors which were used to calculate the stresses and CUFs for the power uprate condition. As a result of its evaluation, the licensee indicated that the maximum calculated stresses are below the Code-allowable limits for both Units 1 and 2. The CUFs provided in Table 6-1 and 6-2 of Attachment 5 to the August 29, 2002, supplemental letter for the limiting SG components are within the code-allowable limit of unity, except for the inspection port bolts which must be replaced every 12 years of service at the uprated power condition.

In addition, the licensee evaluated flow-induced vibration of the U-bend tubes for both Model 44F and Model Delta-47 SGs at Point Beach, Units 1 and 2, respectively. As a result, the licensee concluded that flow-induced vibration of SG tubes will remain within the allowable limits for the MUR power uprate. The NRC staff concurs with the licensee's conclusion since the calculated maximum stress of the tubes due to flow-induced vibration at the power uprate condition is only a small percentage of the material endurance stress.

On the basis of its review, which is described above, the NRC staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the limiting SG components are within the code-allowable limits and are, therefore, acceptable for the proposed 1.4-percent MUR power uprate.

2.2.2.5 Reactor Coolant Pumps

The licensee reviewed the existing design-basis analyses of the Point Beach RCPs to determine the impact of the revised design conditions in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application. Each Point Beach RCP contains a Model 93 single-stage, shaft-seal pump driven by an air-cooled motor. Attachment 5 of the August 29, 2002, supplemental letter indicated that the power uprate evaluation for Point Beach RCPs used the ASME Code, Section III, 1968 edition through the summer 1968 addendum, which is the code of record for Point Beach.

After implementation of the proposed MUR power uprate, the RCS pressure would remain unchanged. As provided in Table 3.4.1-1 of the April 30, 2002, application, the licensee indicated that the design parameter of the RCP temperature (reactor pressure vessel inlet) for the power uprate condition is less than the E-Specification values. Also, there are no significant changes to the design thermal transients. As a result of the evaluation, the licensee concluded that the current Point Beach Model 93 RCPs would remain in compliance with the applicable ASME Code requirements for structural integrity at the proposed power uprate conditions.

On the basis of its review, which is described above, the NRC staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed uprated conditions with a 1.4-percent power increase from the current rated power, would remain in compliance with the requirements of the codes and standards under which Point Beach was originally licensed.

2.2.2.6 Pressurizer

The licensee evaluated the limiting design locations within the pressurizer components (e.g., the pressurizer spray nozzle, the surge nozzle, safety and relief nozzle, lower head well and penetration, support skirt, manway and instrument nozzle, and upper head and shell) for operation at the uprated conditions. The evaluation was performed using the ASME Code, Section III, 1965 Edition, through Summer 1966 addenda, which is the code of record for Point Beach. The key parameters in the current Point Beach Pressurizer Stress Report were compared against the revised design conditions in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application for the proposed MUR power uprate. The proposed MUR power uprate would not change the maximum RCS pressure and the pressurizer temperature, T_{sat}. Therefore, the existing design-basis analyses with the lowest T_{hot} and the lowest T_{cold} conditions, which maximize thermal stresses in the pressurizer components, remain bounding for the proposed MUR power uprate. The calculated CUFs at the uprated condition were found to be below the code-allowable limit of unity as shown in Table 7-1 of Appendix 5 to the August 29, 2002, supplemental letter. As a result of its evaluation, the licensee concluded that the existing pressurizer components would remain adequate for plant operation at the proposed 1.4-percent MUR power uprate, while the RCS pressure would remain unchanged. The NRC staff finds that the existing design-basis analyses, discussed above, bound the proposed 1.4 percent MUR power uprate, and agrees with the licensee's conclusion.

2.2.2.7 Nuclear Steam Supplying System Piping and Pipe Supports

The proposed MUR power uprate for Point Beach involves an increase in the temperature difference across the RCS. The licensee evaluated the NSSS piping and supports by reviewing the design-basis analysis against the uprated power design system parameters, transients, and the LOCA-dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The USAS B31.1 Power Piping Code was used for the power uprate evaluation of RCS piping. The surge line which was evaluated in accordance with requirements of the ASME B&PV Code, Section III, 1986 edition. The NRC staff finds this acceptable in accordance with 10 CFR 50.55a.

In its August 29, 2002, supplemental letter, the licensee provided an evaluation of the primary loop piping and surge line for the effects of the proposed MUR power uprate. The evaluations of the RCS piping and surge line for Point Beach were performed to assess the impact of revised design parameters provided in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application up to a power level of 1650 MWt. The licensee indicated that the WESTDYN computer code was used for the analysis for deadweight, seismic, LOCA, and thermal loads, and load combinations in accordance with the USAS B31.1 Power Piping Code. The results are summarized in Table 4-1 of Enclosure 3 to the August 29, 2002, supplemental letter. The maximum calculated stresses are shown to be less than the code-allowable limits. The stresses and CUF for the surge line are summarized in Table 4-2 of Enclosure 3 to the August 29, 2002, supplemental letter and they are below the code-allowable stress limits and the fatigue CUF limit of 1.0.

The licensee indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles would be unchanged for the Point Beach MUR power uprate. The proposed MUR power uprate would not change the maximum RCS pressure. The design-basis LOCA forces, due to postulated primary loop guillotine breaks, have been eliminated using the loop leak-before-break (LBB) methodology for Point Beach. With the use of LBB technology, LOCA forces for the power uprate condition were derived based on postulation of breaks in three branch lines at the surge line nozzle on the hot leg, the accumulator line nozzle at the cold leg, and the RHR line nozzle on the hot leg. As such, the design-basis LOCA hydraulic forcing functions are bounding for the LOCA loads at the uprated power condition. Furthermore, the deadweight and seismic loads would not be affected by the proposed MUR power uprate. The licensee concluded that the existing stresses, fatigue CUFs and loads remain bounding for the proposed MUR power uprate for the NSSS components, including the RCL piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e., heat exchangers, pumps, valves and tanks). Therefore, these components will continue to be in compliance with the code of record for Point Beach.

On the basis of its review of the licensee's submittals, which is described above, the NRC staff concurs with the licensee's conclusion that the existing NSSS piping and supports, primary equipment nozzles, primary equipment supports, and auxiliary lines connecting to the primary loop piping will remain in conformance with the design-basis criteria, as defined in the Point Beach FSAR, and are, therefore, acceptable for the proposed 1.4-percent MUR power uprate.

2.2.2.8 Balance-of-Plant Systems and Motor-Operated-Valves

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design-basis parameters with the proposed power uprate conditions. The BOP piping systems that were evaluated for the proposed MUR power uprate include the main steam, feedwater, SG blowdown, and auxiliary feedwater systems. The licensee evaluated these affected systems at the uprated power level by comparing the input parameters for the current piping analysis reports against the design parameters in Table 3.4.1-1 of Attachment 1 to the April 30, 2002, application (e.g., RCS temperatures, steam temperature, and steam flow rate) for up to 1650 MWt reactor core power. As a result, the licensee concluded that the existing design-basis analyses for the BOP piping, pipe supports, and components remain bounding for the proposed 1.4-percent MUR power uprate.

The licensee also reviewed the programs, components, structures, and non-NSSS system issues as they relate to the proposed MUR power uprate. In Attachment 1 to the August 29, 2002, supplemental letter, the licensee indicated that its MOV program used the maximum design-basis differential pressure (worst-case scenario) that is expected during normal and emergency operation of MOVs. This maximum operating design system pressure would not change as a result of the 1.4-percent MUR power uprate. Therefore, the licensee concluded that the safety-related MOVs at Point Beach would continue to be capable of performing their intended functions at the uprated power condition.

The licensee reviewed the evaluation of GL 95-07 associated with pressure locking and thermal binding for safety-related gate valves. The licensee found that the existing analysis conditions using the maximum design conditions remain bounding for the 1.4-percent MUR power uprate. The licensee reviewed the evaluation of its GL 96-06 program with respect to the overpressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was performed at 102 percent of 1518.5 MWt and is therefore bounding for the proposed 1.4-percent MUR power uprate. On the basis of the above information, the NRC staff concurs with the licensee's conclusions that the proposed 1.4-percent MUR power uprate would have no adverse effects on the safety-related valves or the conclusions of GL 89-10, GL 95-07, GL 96-05, and GL 96-06.

As a result of the above evaluation, the NRC staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and remain within the design-basis analyses for the proposed 1.4-percent MUR power uprate.

2.2.3 Conclusion

On the basis of its review, the NRC staff concurs with the evaluations performed by the licensee for the NSSS and BOP systems, piping, supports, RPVs, and RVIs. The NRC staff finds the licensee's evaluations, which were performed in accordance with the licensing code of record and the original design basis, are bounding for operation at 1540 MWt. Therefore, the NRC staff concludes that the foregoing components are acceptable for the proposed operations at Point Beach at the uprated power level of 1540 MWt.

2.3 Instrumentation and Controls

2.3.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power and the uncertainties in the calculated values of this thermal power determine the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. Appendix K to 10 CFR Part 50 requires LOCA and emergency core cooling system (ECCS) analyses to assume that the reactor has been operating continuously at a power level at least 102.0 percent of the licensed thermal power to allow for uncertainties, such as instrument error. The phrase "such as" suggests that the 2-percent power margin was originally intended to address uncertainties related to heat sources in addition to the instrument measurement uncertainties. However, the NRC staff has concluded that the 2-percent power margin requirement of the original ECCS rulemaking was solely based on the considerations associated with power measurement uncertainties. This conclusion could justify a reduced margin between the licensed power level and the power level assumed in the ECCS analysis and, therefore, could justify a power uprate.

In order to reduce an unnecessarily burdensome regulatory requirement and to avoid unnecessary exemption requests, the Commission published a final ECCS rule, amending Appendix K in the June 1, 2000, *Federal Register.* As an alternative to maintaining the current 2-percent power margin, the final rule allows licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power. Licensees may apply the reduced margin to operate the plant at a level higher than the current licensed power or use the margin to relax ECCS-related TSs. The final rule, by itself, does not allow licensees to increase the licensed power level without NRC staff approval. Since the maximum power level at which a nuclear power plant may be operated is set forth in a license condition, and is reflected in TS limits, proposals to raise the licensed power level must be reviewed and approved by the NRC staff under the license amendment process. A license amendment request to support a proposed 1.4-percent MUR power uprate should include a justification for the reduced power measurement uncertainty.

Caldon Topical Report ER-80P provides calculations for thermal power measurement uncertainties using Chordal LEFM flow and temperature measurements applicable to a two-loop PWR or a two-feedwater-line boiling-water reactor (BWR). Based on this calculation, the summary of the report stated that the LEFM is accurate to ±0.6 percent of thermal power at a 95-percent confidence level, versus ±1.4 percent for current instrumentation. The NRC staff approved this Caldon Topical Report ER-80P for a 1-percent power uprate and for exempting plants which used Caldon's LEFM from the Appendix K requirement. The NRC staff also approved Caldon Topical Report ER-160P (Reference 8), a supplement to ER-80P, for a power uprate to 1.4 percent, by safety evaluation report dated January 19, 2001. The licensee used these topical reports as the basis for justifying the license amendment request.

2.3.2 Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant's NSSS. This calculation is called the "secondary calorimetric" for a PWR and the "heat balance" for a BWR. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow and

feedwater net enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature would result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

Instruments for measuring feedwater flow typically use an orifice plate, a venturi meter, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. Of the three differential pressure devices, a venturi meter is most widely used for feedwater flow measurement in nuclear power plants. The feedwater temperature is typically measured by resistance temperature detectors (RTDs). The Point Beach design uses a venturi for flow and RTDs for temperature measurement in each of the SG feedwater systems. The major advantage of the venturi meter is a relatively low head-loss as the fluid passes through the device. The major disadvantage of the device is fouling, which causes the meter to indicate a higher differential pressure, and hence, a higher-than-actual flow rate. This leads the plant operator to calibrate nuclear instrumentation high. Calibrating the nuclear instrumentation high is conservative with respect to reactor safety, but causes the electrical output to be proportionally low when the plant is operated at its thermal power rating. To eliminate the fouling effects, the flow device has to be removed, cleaned, and recalibrated. Due to the high cost of recalibration, the industry assessed other flow measurement techniques and found the LEFM to be a viable means of reducing flow measurement uncertainty.

The Caldon Chordal LEFM is an ultrasonic flow meter, using acoustic energy pulses to determine the feedwater mass flow rate and temperature. The meter is based on transit time technology (also called the “time-of-flight” or “counter-propogation” technology). The transit time technology sends an ultrasonic signal diagonally through the fluid and then measures the time the signal takes to travel upstream and downstream. The sound travels faster when the pulse traverses the pipe with the flow, and slower when the pulse traverses the pipe against the flow. The difference in these times is proportional to the velocity of the fluid in the pipe. The LEFM uses these transit times and the time differences between pulses to determine the fluid velocity and temperature (temperature of pure water can be determined from its sound velocity and pressure). The LEFM system uses eight transducers in a configuration of two transducers on each of the four acoustic measurement paths in a single plane of the spool piece.

At Point Beach, the LEFM system consists of an electronic cabinet in the process control area of the plant and a measurement section (a spool piece) that was installed in the early 1980s in the 20-inch main feedwater header. In 1995, Caldon replaced the old electronics of the LEFM with a new algorithm to reduce uncertainty. Instead of testing the spool piece and the new electronics in a laboratory in a plant-specific configuration, MPR Associates performed an uncertainty analysis using path velocity measurement data from the new system. MPR Associates documented its analysis in Report No. MPR-1619, and the LEFM measurement uncertainty calculated in this report is used in the plant-specific reactor thermal power uncertainty calculation provided in the Westinghouse Revised Thermal Design Procedures (RTDPs). The LEFM is a single digital system controlled by software using the ultrasonic transit time method to measure four line-integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the four measured velocities to determine the mass flow rate and the fluid temperature. These measurements are used by the plant’s process computer system (PPCS) to determine the reactor thermal output.

The licensee stated that although the system's function is not safety-related (providing flow and temperature inputs only to the calorimetric calculation), the system's software was developed and would be maintained under a verification and validation (V&V) program that is in compliance with Institute of Electrical and Electronic Engineers (IEEE) Standard 7-4.3.2-1993, Annex E, and ASME Standard NQA-1-1999, Addenda Subpart 2.7. The V&V program is also consistent with the guidance for software V&V in the Electric Research Power Institute (EPRI) Report TR-103291, "Handbook for Verification and Validation of Digital Systems," dated December 1994. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The LEFM indications of feedwater flow and temperature would be displayed on the local display panel and transmitted to the PPCS. This information would be directly substituted for the venturi-based flow indications and the RTD temperature indications currently used in the plant calorimetric calculation. The venturi-based feedwater flow measurement would continue to be used for feedwater control and other functions, as currently used. A real-time display of thermal power using LEFM will be available in the main control room and an audible alarm will annunciate to the operator when the LEFM is not operating within its design-basis accuracy. The licensee stated that the LEFM provides an online verification of the accuracy of the feedwater flow and temperature and would significantly improve measurement accuracy and reliability.

Caldon Topical Report ER-80P and its supplement, ER-160P, describe the LEFM system and include calculations of thermal power uncertainties for a typical two-loop PWR or BWR using the LEFM Check system for feedwater flow and temperature measurement. The calculation results for a typical PWR or BWR showed a total thermal power determination uncertainty of ±0.6 percent with a 95-percent confidence level. The report provides a generic basis for the proposed 1.4-percent MUR power uprate and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties.

The NRC staff's March 8, 1999, safety evaluation report approving Caldon Topical Report ER-80P included four additional items to be addressed by a licensee requesting a power uprate. The licensee's submittals addressed each of the four items as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

The licensee addressed this item in its April 30, 2002, application. The licensee stated that implementation of the power uprate would include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level and would be based on Caldon's recommendations. Caldon will notify the licensee of any deficiency that could affect the design-basis accuracy of the LEFM. Point Beach has a corrective action program for receiving and addressing vendors' deficiency reports, reporting deficiencies to the vendor, and performing the necessary and recommended corrective actions. The LEFM software would be maintained under Caldon's V&V program, and all other instrumentation affecting the power calorimetric, including the PPCS, would continue to be maintained in accordance with the existing plant maintenance and calibration procedures. The LEFM operability requirements would be included in the Point Beach Technical Requirements Manual (TRM). If the LEFM is not operable, the plant may continue operating at 1540 MWt and may use the LEFM- corrected venturi core thermal power output calculation prior to the next

performance of TS surveillance requirement (SR) 3.3.1.2. If the LEFM is not returned to service prior to the next performance of TS SR 3.3.1.2, TS 5.6.4(b) requires that a power measurement uncertainty consistent with the instruments used be applied. In such an instance, the TRM would provide that the licensee reduce power to 1520 MWt, which is consistent with the calculated power measurement uncertainty associated with the feedwater venturies (calculated by WCAP-14787). The licensee would maintain the lower power level until the LEFM is returned to operable status.

(2) For plants that currently have LEFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Point Beach currently uses LEFM spool pieces for correcting the venturi fouling effects. The new improved LEFM is a replacement and would be bounded by the assumptions and analysis set forth in topical report ER-80P and its supplement, ER-160P.

(3) The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feed water instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

The licensee referenced Revision 1 of two Westinghouse Licensing Topical Reports, WCAP-14787 and WCAP-14788. These two topical reports document the Westinghouse Revised Thermal Design Procedures (RTDPs) and include the plant-specific power calorimetric measurement uncertainty calculations for Point Beach. The calculation methodology conforms to the recommendations of American National Standard Institute (ANSI)/Instrument Society of America (ISA) Standard 67.04, 1994 and Regulatory Guide (RG) 1.105, Revision 2. The NRC staff reviewed these WCAPs during the Fuel Upgrade Project and approved them by a safety evaluation report dated February 8, 2002. However, the NRC staff was still concerned about the methodology, and requested the licensee to confirm that the methodology meets the criteria identified in ISA Standard S67.04, 1994 and RG 1.105, Revision 2.

By its supplemental letter dated August 29, 2002, the licensee confirmed that the methodology meets the above standard and RG and also addressed the NRC staff's concern about the uncertainty values used in the calculations. This methodology has been reviewed and approved by the NRC staff for Westinghouse PWRs. In these calculations, Westinghouse statistically combined the LEFM uncertainty with other instrumentation uncertainties affecting the plant power calorimetric uncertainty. The resulting power calorimetric measurement uncertainty for each Point Beach unit was found to be 0.6 percent of the rated thermal power, which justifies the proposed 1.4-percent MUR power uprate. The NRC staff found that the licensee's calculation is based on an NRC-approved plant setpoint methodology and is, therefore, acceptable.

(4) Licensees of plants where the ultrasonic meter (including the LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

In its August 29, 2002, supplemental letter, the licensee stated that the Point Beach spool pieces were not calibrated in a site-specific hydraulic geometry when they were installed in the early 1980s. After installing new electronics in 1995, MPR Associates performed feedwater flow measurement uncertainty calculations for Point Beach. MPR Associates statistically developed the profile factors and the profile factor uncertainties for the spool pieces using path velocity measurement data provided by the LEFM with replacement electronics. The statistical approach used by MPR Associates as documented in report MPR-1619 is the same approach discussed in Caldon Topical Report ER-80P. Caldon has reviewed and accepted MPR report MPR-1619 for the new LEFM system at Point Beach. The NRC staff accepted the statistical approach during its review of Caldon Topical Report ER-80P. Based on this, the NRC staff finds that the licensee has adequately addressed this issue.

The NRC staff finds that the licensee's response has sufficiently resolved the plant-specific concerns regarding maintenance and calibration of the LEFM and other instrumentation affecting the power calorimetric, the hydraulic configuration of the installed LEFM, processes and contingencies for an inoperable LEFM, and the methodology for calculating the LEFM measurement uncertainty and the plant power calorimetric uncertainty.

2.3.3 Conclusion

Based on the foregoing, which documents the NRC staff's review of the licensee's submittals on the LEFM system and plant power calorimetric uncertainty, the NRC staff finds that the Point Beach thermal power measurement uncertainty, with the LEFM system in Units 1 and 2, is limited to ±0.6 percent of the reactor thermal power and can support the proposed 1.4-percent MUR power uprate. The NRC staff also finds that the licensee adequately addressed the four additional items outlined in the NRC staff's March 8, 1999, safety evaluation report that approved Caldon Topical Report ER-180P.

2.4 Electrical Engineering

2.4.1 Regulatory Evaluation

Prior to the introduction of General Design Criterion (GDC)-17, "Electric Power Systems," of Appendix A to 10 CFR Part 50, 10 CFR 50.34(a)(3)(i) required applicants for construction permits to submit principle design criteria. The principle design criteria for Point Beach are referred to as the "Point Beach GDCs," and documented in the Point Beach FSAR. Point Beach GDC-39 was modeled on draft GDC-39 as proposed in a rulemaking published in the

Federal Register on July 11, 1967, and was used to evaluate the adequacy of the Point Beach electric power systems. Point Beach GDC-39 provides that sufficient offsite and redundant, independent and testable standby auxiliary sources of electrical power are provided to attain a prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. Point Beach GDC-39 provides further that the capacity of the power sources is adequate to accomplish all required engineered safety features functions under all postulated design-basis accident (DBA) conditions.

The regulation at 10 CFR 50.63 requires that all nuclear power plants have the capability to withstand a loss of all AC power for an established period of time, and to recover therefrom. The plant was previously evaluated for SBO.

The regulation at 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety. Under the rule, each licensee must (1) prepare and maintain a record of qualification to document that each item of equipment subject to the rule is qualified for its application, and (2) meets its specified performance requirements when subjected to the environmental conditions predicted to be present when it must perform its safety function up to the end of qualified life.

The main generator is rated at 582 MVA at a 0.9 power factor. The maximum equipment design rating of the main generator is 608 MVA at a 0.92 power factor. The station power output is generated at 19 kV, and is fed through an isolated phase bus to 609 MVA main transformer where it is stepped up to 345 kV transmission voltage and delivered to the switchyard. The design of the system is such that sufficient independence between the various sources of electrical power is provided in order to guard against concurrent loss of all auxiliary power. The station distribution system consists of various auxiliary electrical systems to provide electrical power during all modes of operation and shutdown conditions. The electrical distribution system was previously evaluated to conform to Point Beach GDC-39.

2.4.2 Technical Evaluation

2.4.2.1 Grid Stability

The American Transmission Company (ATC) assessed grid stability and thermal loading for Point Beach. The daily average is 80 MVAR out. The impact study performed by ATC identified no stability issues for the proposed 1.4-percent MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded that the proposed MUR power uprate would not adversely affect the grid stability for the uprate condition and the plant would be operated in the procedural limits of MVAR (100 MVAR in and 200 MVAR out).

2.4.2.2 Main Generator

In its June 26, 2002, supplemental letter, the licensee states that the existing design and procedure limits of the main generator are 582 MVA at 0.924 power factor (pf). The anticipated power uprate of the main generator is 588.2 MVA at 0.927 pf, and the maximum main generator design rating is 608.4 MVA at 0.92 pf.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 588.2 MVA is below the maximum main generator design rating of 608.4 MVA and, therefore, operating the main generator at the uprated power condition is acceptable.

2.4.2.3 Main Transformer

The existing design and procedure limit of the main transformer is 582 MVA. The anticipated power uprate of the main transformer is 588.2 MVA and the maximum main transformer design rating is 609.3 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 588.2 MVA is below the maximum main transformer design rating of 609.3 MVA and, therefore, operating the main transformer at the uprated power condition is acceptable.

2.3.2.4 Isophase Bus

The existing design and procedure limit of the isophase bus is 17,685 amps. The anticipated power uprate of the isophase bus is 17,874 amps and the maximum isophase bus design rating is 20,000 amps.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 17,874 amps is below the maximum isophase bus design rating of 20,000 amps and, therefore, operating the isophase bus at the uprated power condition is acceptable.

2.4.2.5 Station Auxiliary Transformers/Unit Auxiliary Transformers

The existing design and procedure limit of the station auxiliary transformer (SAT)/unit auxiliary transformer (UAT) is 25.2 MVA. The anticipated power uprate of the SAT/UAT would stay the same as 25.2 MVA and the maximum SAT/UAT design rating would be 37.3 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the anticipated power uprate of 25.2 MVA is below the maximum SAT/UAT design rating of 37.3 MVA and, therefore, operating the SAT/UAT at the uprated power condition is acceptable.

2.4.2.6 Motor-Driven Pumps

The existing brake horse power (BHP) on the main feedwater pump motor and the condensate pump are 4,400 BHP and 1,140 BHP respectively. With the proposed 1.4-percent MUR power uprate, the BHP loads on the main feedwater pump motor and the condensate pump are 4,500 BHP and 1,150 BHP respectively. These loads are below the design rating of the main feedwater pump motor and the condensate pump of 5,000 BHP and 1,250 BHP respectively, and the design is therefore acceptable.

2.4.2.7 Emergency Diesel Generators

There is no change to the safety-related loads at uprate conditions and, therefore, the emergency diesel generators would not be affected by the power uprate, and can perform their safety-related functions during a loss of offsite power/LOCA.

The NRC staff's review determined that the proposed 1.4-percent MUR power uprate would not affect the loading on the emergency diesel generator. Therefore, the licensee would continue to meet Point Beach GDC-39 requirement with the proposed 1.4-percent MUR power uprate.

2.4.2.8 Station Blackout

The only potential impact of the proposed 1.4-percent MUR power uprate on the ability of the plant to withstand and recover from an SBO is the increased decay heat that must be removed from the RCS. The methodology and the assumptions associated with the SBO analysis with regard to equipment operability would not be changed with the proposed uprate. There would be no change in the ability of the turbine-driven auxiliary feedwater pumps, supplied with steam from the SGs, to support reactor heat removal due to the uprate. The TS minimum required volume in the condensate storage tank is 13,000 gallons. This volume would remain acceptable for the proposed 1.4-percent MUR power uprate since the calculation took into account a 4.5-percent uncertainty on the initial power level. This uncertainty on power bounds the uprate to 1540 MWt. Therefore, the ability of the plant to respond to an SBO would not be altered due to the MUR power uprate.

The NRC staff reviewed the licensee's submittal and concluded, based on the above, that the uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event for the uprate condition.

2.4.2.9 Environmental Qualification of Electrical Equipment

Electrical equipment specified in 10 CFR 50.49 is qualified based on normal and accident environmental conditions in the containment and portions of the auxiliary and turbine buildings. The licensee evaluated the changes for normal and accident operating conditions for a 1678 MWt extended power uprate. The evaluations bound the proposed 1.4-percent MUR power uprate.

The licensee analyzed the impact of a power uprate as high as 10.5 percent on the normal design temperatures and the environmental conditions in containment, the auxiliary building and the turbine building. The normal conditions in these areas would not be impacted and are bounded by the current design. The impact of the uprate on normal operational doses to EQ equipment has been evaluated. The dose associated with the power uprate is bounded for the uprated conditions by existing analysis. Exposures in areas inside and outside of containment that are directly exposed to radiation considered in current qualification documentation would remain bounding for power uprated conditions.

Accidents causing the most severe environmental conditions include LOCAs and main steamline breaks (MSLB) inside containment, and high energy line breaks outside containment. The MSLB for inside containment bounds the condition for a 1.4 percent power uprate since the analyses are performed at 102 percent of current power. Mass and energy releases for a steamline break outside containment are taken from a Westinghouse source and remain conservative for the uprated conditions. The LOCA containment response is currently analyzed at 102 percent of current power. Therefore, the containment analysis does not change for the power uprate and there are no changes in temperature, pressure, and humidity following a LOCA.

Based on its evaluation, the NRC staff concluded that the proposed MUR power uprate would have no adverse impact on the pressure, temperature, and radiation environments used in EQ analyses, as set forth above, and the plant would meet the requirements of 10 CFR 50.49 at the power uprated conditions

2.4.3 Conclusion

The NRC staff has evaluated the effect of MUR power uprate on the necessary electrical systems and EQ of electrical components. Results of these evaluations show that the increase in core thermal power would have negligible impact on grid stability, SBO, or the EQ of electrical components. This is consistent with Point Beach GDC-39, 10 CFR 50.63, and 10 CFR 50.49 and the proposed change is, therefore, acceptable.

2.5 Structural Integrity and Metallurgy Section

2.5.1 Regulatory Evaluation

This section identifies those NRC regulations, regulatory guides, NRC staff positions, and other guidance documents used by the NRC staff in the review of the applicable sections of the licensee's submittal.

Regarding RPV integrity issues addressed in Section 3.4.4 of Attachment 1 to the licensee's April 30, 2002, application, the NRC staff reviewed information submitted by the licensee pertaining to the development of RPV P-T limit curves, upper shelf energy (USE) analyses, the RPV material surveillance programs, and the evaluation of the susceptibility of the Point Beach RPVs to failure during a pressurize thermal shock (PTS) event.

The NRC's regulatory requirements related to the establishment of RPV P-T limit curves for any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests and requirements related to RPV material USE are given in Appendix G to 10 CFR Part 50. With regard to the development of RPV P-T limits, Appendix G to 10 CFR Part 50 also references the requirements given in the ASME B&PV Code, Section XI, Appendix G, which are incorporated into the Commission's regulations by reference in 10 CFR 50.55a. Additional guidance for the NRC staff's review of RPV P-T limit curves and USE analyses is provided in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," SRP Section 5.3.2, "Pressure-Temperature Limits," and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." Appendix K to Section XI of the ASME Code and Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb" may also be used as guidance when USE-equivalent margins analyses are required.

The NRC's regulatory requirements related to the establishment of a plant's RPV surveillance capsule program and withdrawal schedule are given in Appendix H to 10 CFR Part 50, which also references the guidance in American Society for Testing and Materials (ASTM) Standard Practice E 185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." SRP Section 5.3.1, "Reactor Vessel Materials," also applies.

The NRC's regulatory requirements regarding design criteria to ensure that facility RPVs are protected from failure during a PTS events are given in 10 CFR 50.61. As with P-T limits, Branch Technical Position MTEB 5-2 may also apply with respect to the determination of initial, unirradiated properties of RPV materials.

Regarding Section 3.4.2 of Attachment 1 to the licensee's April 30, 2002, letter, the NRC staff reviewed the information pertaining to the effect of the requested MUR power uprate on the structural integrity of the reactor vessel internals. Maintenance of the structural integrity of the RPV internals is required in order to demonstrate that the functional requirements of the RPV internals are met. These functional requirements include core support and emergency core cooling system (ECCS) performance aspects. As such, the structural integrity of the RPV internals is linked to regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry. Additional guidance regarding the evaluation of the structural integrity of RPV internals may be found in SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," and in ASME Code Sections III and XI, or other standards which were used in the NRC's review of the original licensing basis of a particular facility.

2.5.2 Technical Evaluation

In Section 3.4.4 of Attachment 1 to its April 30, 2002, application, the licensee made three specific conclusions regarding the RPV integrity evaluations required by Appendix G to 10 CFR Part 50 and 10 CFR 50.61. In the first conclusion regarding the issue of RPV beltline material USE, the licensee stated:

> Since the PBNP [Point Beach Nuclear Plant] upper shelf energy (USE) value at the end of life (EOL) in the limiting welds was predicted to fall below the NRC (10 CFR 50) Appendix G requirements of 50 ft-lb, a fracture mechanics evaluation was performed to demonstrate acceptable equivalent margins of safety against fracture .... This analysis showed that the limiting PBNP welds satisfied the requirements of the ASME Code, Section XI, Appendix K, for ductile flaw extension and tensile instability by at least a margin of 50 percent .... As outlined above, the 1.4 percent power uprate has no adverse impact upon USE evaluations.

In order to independently evaluate the licensee's conclusion, the NRC staff requested additional information, which was supplied by the licensee in its August 29 and October 3, 2002, supplemental letters. This additional information included: (1) more detailed information regarding the equivalent margins analysis (EMA) conducted by the licensee and, (2) conservative RPV fluence values for the limiting Point Beach materials at 34 effective full power years (EFPY) of operation, consistent with the projected end of the current Point Beach operating licenses. In this case, the fluence values submitted by the licensee are characterized as "conservative" since the licensee considered not only the current proposed 1.4-percent MUR power uprate, but also a potential extended power uprate to 1678 MWt (approximately an additional 9 percent) which the licensee is considering as the subject of a future submittal.

Based on this information, and information submitted previously regarding the copper content of the Point Beach RPV materials, the NRC staff independently evaluated the limiting materials with respect to end of license USE for the Point Beach RPVs. The NRC staff's independent

analysis showed that the cumulative usage factors remained below 1.0 for both RPVs. Based on the NRC staff's analysis, the NRC staff agrees with the licensee's conclusion that the 1.4-percent MUR power uprate has no adverse impact upon USE evaluations since, in the words of Appendix G to 10 CFR Part 50, "margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code" were demonstrated.

In the second RPV integrity conclusion in Section 3.4.4 regarding the existing P-T limit curves for Point Beach, the licensee stated:

> The proposed power uprate will slightly change the rate of neutron flux to the reactor vessel. The P-T curves [which are contained in a licensee-controlled pressure-temperature limit report (PTLR)] will be revised using the existing processes prior to exceeding the above stated fluence and EFPY values. This is monitored through review of the EFPY values in the monthly operating reports. The PBNP P-T curves are scheduled for revision prior to January 2004, which would allow [power uprate] operation without exceeding the limiting EFPY values. Therefore, the proposed 1.4 percent power uprate has no adverse impact upon existing P-T curves.

As noted by the licensee, the NRC has previously approved Point Beach to implement a PTLR methodology for the development and control of the Point Beach P-T limit curves. The requirement that the licensee implement a specific NRC-approved methodology for the development of the plant's PTLR is contained in Point Beach TS Section 5.6.5. Changes in plant operation associated with the proposed 1.4-percent MUR power uprate would affect input values (i.e., the end of license RPV fluence values for the Point Beach RPVs) for the PTLR methodology, but would not compromise the acceptability of the PTLR methodology itself. Hence, the NRC staff concludes that the licensee will, in keeping with the requirements of its PTLR methodology as referenced in the facility TSs, update the P-T limit curves as appropriate.

In the third RPV integrity evaluation conclusion in Section 3.4.4 regarding the evaluation of PTS, the licensee stated:

> The limiting RT_{PTS} [pressurized thermal shock reference temperature] value for both units is the intermediate to lower shell circumferential weld. The 10 CFR 50.61 limit for circumferential welds is 300 °F. The Unit 1 weld (SA-1101) is calculated [to have an RT_{PTS} value of 276 °F, and the Unit 2 weld (SA-1484) is predicted at 292 °F. The RT_{PTS} values listed are for EOL with an extended power uprate to 1678 MWt. EOL conditions were defined as 32 EFPY (Unit 1) and 34 EFPY (Unit 2). These values bound the 1.4 percent [power uprate]. PBNP RT_{PTS} values remain within regulatory limits following the 1.4 percent [power uprate].

In its August 29, 2002, supplemental letter, the licensee provided the actual projected fluence values at the inside surface of the limiting RPV welds at end of license conditions, including the conservative assumption of an extended power uprate to 1678 MWt. Based on this updated end of license fluence information, and information submitted previously regarding the copper content of the Point Beach RPV materials, the NRC staff independently evaluated the limiting materials with respect to end of license RT_{PTS} values. Based on the NRC staff's analysis, the NRC staff agrees with the licensee's conclusion that the RT_{PTS} values for the materials in the Point Beach RPVs would remain below the PTS screening criteria established in 10 CFR 50.61.

Hence, with regard to the requirements of 10 CFR 50.61, operation of the Point Beach RPVs would remain acceptable through the end of the current facility operating licenses.

In the final conclusion in Section 3.4.4 regarding the Point Beach RPV surveillance program, the licensee stated:

> [t]he original PBNP surveillance program was prepared in accordance with ASTM E 185-66 and consisted of six surveillance capsules in each unit....All surveillance capsules (except for standby capsules) have been removed and tested.
>
> The actual heats of the limiting metals were not included in the PBNP capsules. Therefore, PBNP is a member in the B&W [Babcock & Wilcox] Owners Group Materials Subcommittee that has allowed access to irradiated surveillance data of all limiting welds. PBNP and the B&W Owners Group have an ongoing program for removal and testing of all PBNP limiting weld materials as part of the PBNP surveillance capsule program.

Since all RPV surveillance capsules required to be tested under the licensee RPV surveillance program, established in accordance with ASTM E 185-66 and Appendix H to 10 CFR Part 50, have already been removed and tested, changes resulting from the proposed 1.4-percent MUR power uprate cannot affect the facility's compliance with the regulations and its current licensing basis. Further, changes resulting from the proposed 1.4-percent MUR power uprate would have no effect on data that the licensee will acquire from its interaction with the B&W Owners Group Materials Subcommittee. Therefore, the NRC staff concludes that the RPV surveillance programs for Point Beach would continue to comply with the requirements of Appendix H to 10 CFR Part 50 after implementation of the proposed 1.4-percent MUR power uprate.

The final topic reviewed by the NRC staff was the licensee's determination regarding the continued integrity of the Point Beach RPV core support structures and vessel internals, addressed in Section 3.4.2 of Attachment 1 to the licensee's April 30, 2002, application. The licensee noted that as part of the Point Beach Replacement Steam Generator Program, the RPV core support structures and vessel internals had been reevaluated for operation up to 1650 MWt. The licensee noted that although these components had been designed and installed prior to the introduction of ASME Code, Section III, Subsection NG (which addresses RPV core support structures), the licensee utilized analysis procedures similar to those addressed in that ASME Code subsection in order to perform the reevaluation. The conclusions of the licensee's analysis were that the RPV core support structures and vessel internal would maintain their integrity under conditions consistent with operation up to 1650 MWt.

Based on the changes in pressure, temperature, and flow loads expected to result from the proposed 1.4-percent MUR power uprate, as described above, the NRC staff agrees with the licensee's conclusion that the integrity of the RPV core support structures and vessel internals would be maintained such that the licensee's ability to meet the regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry would not be adversely impacted.

2.5.4 Conclusion

The NRC staff has reviewed the information provided in Sections 3.4.2 and 3.4.4 of Attachment 1 to the licensee's April 30, 2002, application, and the additional information provided by the licensee in its August 29 and October 3, 2002, supplemental letters. The NRC staff has concluded that sufficient information regarding the continued qualification of the Point Beach RPVs, core support structures, and vessel internals has been provided to support NRC staff's approval of the proposed 1.4-percent MUR power uprate for Point Beach.

2.6 Component Integrity

2.6.1 Regulatory Evaluation

The NRC staff used the following documents in performing its evaluation of the licensee's proposed 1.4-percent MUR power uprate: (1) NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," (2) GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," and (3) EPRI report NSAC-202L-R2, "Recommendation for Effective Flow-Accelerated Corrosion Program."

2.6.2 Technical Evaluation

2.6.2.1 Flow-Accelerated Corrosion

Flow-accelerated corrosion (FAC) occurs in components made of carbon and low-alloy steel exposed to flowing water or wet steam environments. The wear rates caused by FAC depend on several operating parameters. The most important are: velocity of single-phase or two-phase flow, temperature, water chemistry, and the geometry of the affected components. The increase in reactor power after the power uprate would affect the flow velocities and temperatures in various systems in the plant and could result in increased wear rates in the components susceptible to FAC.

In its August 29, 2002, supplemental letter responding to NRC staff RAIs, the licensee stated that a power uprate of the Point Beach plant would produce only a very small change in fluid temperature and its effect on the wear rates due to FAC would be insignificant. However, velocity changes in some systems would be sufficiently high to produce a measurable increase in the wear rates. The most significant increase in flow velocities would occur in the feedwater and condensate systems. Therefore, it is expected that in these systems, the most noticeable wear rates in FAC-susceptible components would occur.

The licensee has determined that the highest wear rates would occur in a small portion of the condensate system, where they will increase by 7.5 percent. However, in the feedwater system carrying high energy water, the power uprate would increase water velocity by 1.2 percent and the corresponding wear rates by less than 1 percent. The licensee has indicated that Point Beach has a program for predicting wear rates due to FAC that is based on the EPRI-developed CHECWORKS code. This program would remain in place following the implementation of the proposed power uprate and will allow the licensee to make future wear-rate assessments. The NRC staff finds this acceptable because it will prevent future failure of components by FAC.

2.6.2.2 Steam Generators

Point Beach Unit 1 has four Westinghouse model 44F SGs with Alloy 600 thermally treated tubing. Unit 2 has four Westinghouse model Delta 47 SGs with Alloy 690 tubing. The licensee stated that the analysis for the power uprate to 1540 MWt will allow SG tube plugging (SGTP) in the range from 0 percent - 10 percent maximum in any SG. To assess the licensee's evaluation of SG structural and leakage integrity, the NRC staff reviewed the effect, if any, that the power uprate would have on SG tube degradation including condition monitoring and operational assessments applicable to Units 1 and 2.

According to the licensee, changes in the reactor vessel outlet temperature (T_{HOT}) and the secondary-side pressure are key parameters in determining corrosion effects. These parameters are inputs into calculations used to determine tube integrity.

Since the licensee only proposed a 1.4-percent MUR power uprate for Point Beach, the licensee intends to maintain the same T_{hot} (the hot leg temperature) range before and after power uprate. Industry experience has shown that, in general, a high T_{hot} correlates with increased tube degradation. Therefore, limiting T_{hot} to the pre-uprate range should prevent the rate of overall tube degradation from increasing after the power uprate.

Experience with power uprates at other plants has shown that a significant increase in steam flow (>5 percent) and a significant decrease in steam pressure (>100 psi) may affect flow-induced tube vibration and result in increased anti-vibration bar (AVB) wear. However, the 1.4-percent MUR power uprate slightly increases the steam flow rate and slightly decreases the steam pressure. The licensee concluded that the 1.4-percent MUR power uprate would have a negligible impact on the projected AVB wear rate and would not significantly impact future tube wear at the AVBs. Based on its review, the NRC staff agrees with the licensee's conclusion and finds it acceptable

The licensee's current SG program includes the identification and disposition of loose parts either by removal or monitoring. In addition, the SG program provides for the evaluation of the impact of loose parts through condition monitoring and operational assessments. Therefore, the NRC staff concludes that, based on the slight increase of the steam flow rate and slight decrease of the steam pressure, the 1.4-percent MUR power uprate would have a negligible impact on tube wear caused by loose parts, and that the licensee has provided reasonable assurance that challenges to SG tube integrity from secondary-side loose parts would be managed via the current site SG inspection program.

The licensee stated that the minimal change in the secondary-side pressure would not produce a quantifiable impact on degradation rates or structural and leakage integrity. In addition, the licensee would disposition new degradation (unanticipated degradation) through the plant corrective action program and the performance of a root cause analysis.

By letter dated July 9, 2002, the NRC staff requested additional information regarding active degradation mechanisms at Point Beach. The NRC staff requested the licensee to provide a summary of its operational assessment for any active degradation mechanisms under power uprated conditions. In its response dated August 29, 2002, the licensee stated that there are no active degradation mechanisms in Point Beach Unit 1 or Unit 2 SGs. The licensee stated

that the SGs would continue to be assessed for degradation per site directives that meet the EPRI guidelines.

Based on a structural evaluation of RCS components, the licensee concluded that operation at uprated conditions was bounded by RCS design conditions and would not impact the stress or fatigue of any RCS components, including the SGs.

Based on the information the licensee provided, as discussed above, the NRC staff agrees with the licensee's conclusion that the power uprate would not have a significant impact on its SG tube structural and leakage integrity.

2.6.3 Conclusion

As set forth above, the NRC staff has evaluated the licensee's programs for managing the effects of a 1.4-percent MUR power uprate on the corrosion wear rates due to FAC and on the performance of the SGs at Point Beach. On the basis of its evaluation, the NRC staff finds that in both of these cases, the effects would be very small and the licensee has the procedures in place to account for them.

2.7 Dose Consequences Analysis

2.7.1 Regulatory Evaluation

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

As with any license amendment, the licensee must show, and the NRC staff must find, that the plant would continue to meet the dose limit criteria given in 10 CFR Part 100, and Point Beach GDC-70. The NRC staff considered whether the amendment conformed to the guidance in applicable sections of SRP Chapter 15 for DBAs. Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," recommends that to improve efficiency of the NRC staff's review, licensees requesting an MUR power uprate should identify existing DBA analyses of record which bound plant operation at the proposed uprated power level. For any DBA for which the existing analyses of record do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

2.7.2 Technical Evaluation

The NRC staff reviewed the impact of the proposed changes on DBA radiological analyses, as documented in Chapter 14 of the Point Beach FSAR. The NRC staff used the information currently contained in the Point Beach FSAR in addition to that submitted by the licensee in support of the requested license amendments. The licensee identified the existing DBA

radiological analyses of record that bound the conditions expected at the proposed uprated power level. The licensee stated that the FSAR Chapter 14 non-LOCA DBA radiological analyses were performed at a core power level of 1650 MWt, with the exception of the fuel handling accident, which was performed at 1518.5 MWt with a 2-percent uncertainty, for a total power level of 1549 MWt. The existing Chapter 14 LOCA DBA radiological analysis was also performed at a power level of 1549 MWt. The NRC staff verified that the existing Point Beach FSAR Chapter 14 radiological analyses bound the proposed 1.4-percent MUR power uprate conditions.

The licensee identified and discussed one existing FSAR Chapter 14 radiological analysis that does not bound the proposed uprated power: FSAR Chapter 14.2.3, "Accidental Release -- Waste Gas." FSAR Chapter 14.2.3 describes the analyses of accidental releases of waste gas, which include the rupture of the gas decay tank, the volume control tank, the charcoal-filled gas decay tank and the cryogenic absorber vessel. The input assumptions for these existing analyses are associated with a core power of 1518.5 MWt.

The waste gas storage system removes radioactive gases from the RCS. For the FSAR analyses, the licensee assumed the activity within the waste gas storage system is the maximum amount that could accumulate from operation with cladding defects in 1 percent of the fuel elements. This is much higher than the actual or expected number of defective fuel elements.

The radioactive gas activity in the reactor coolant is directly proportional to the core thermal power. In the submittal, the licensee evaluated the impact of the proposed 1.4-percent MUR power uprate by multiplying the existing FSAR site boundary dose values for each accident by a conservative scaling factor of 11 percent, which corresponds to an extended power uprate. This bounds the 1.4-percent MUR power uprate. These scaled dose values remain well below the 10 CFR Part 100 dose limits.

The NRC staff finds this evaluation to be acceptable because of the conservatism both in the original FSAR analyses and the use of the 11 percent scaling factor. Additionally, the licensee stated in the submittal that a formal engineering evaluation of the effect of the power uprate on the FSAR 14.2.3 accidents would be documented prior to the implementation of the proposed 1.4-percent MUR power uprate and the results incorporated into the next FSAR update using 10 CFR 50.59.

2.7.3 Conclusion

Based on the above discussion, the NRC staff finds that the existing FSAR Chapter 14 radiological analyses, which were performed at 102 percent or more of the current rated core thermal power of 1518.5 MWt remain bounding for the proposed 1.4-percent MUR power uprate to 1540 MWt, considering the higher accuracy of the Caldon LEFM flow meters. These analyses, which have been previously reviewed and found acceptable by the NRC staff, show that the radiological consequences of postulated DBAs meet the dose limits given in 10 CFR Part 100 and Point Beach GDC-70. The NRC staff finds that the licensee's evaluation of the accidental release of waste gas is acceptable. The licensee has demonstrated that the radiological consequences of a rupture in the waste gas decay tank, the volume control tank, the charcoal-filled decay tank, or the cryogenic absorber vessel would remain considerably

below the 10 CFR Part 100 limits. Therefore, the NRC staff finds that the proposed changes are acceptable with respect to the radiological consequences of DBAs.

2.8 Containment Analyses

The licensee indicated that no changes to the containment structure or containment isolation systems are needed as part of the proposed 1.4-percent MUR power uprate. The licensee further indicated that systems are periodically tested for containment design integrity. The licensee further indicated that there would be no changes in the test programs based on the proposed 1.4-percent MUR power uprate.

The licensee indicated that it evaluated the containment ventilation system for normal and accident operations to determine if the system would need modification as a result of additional heat loads at a full 10.5-percent power uprate. The licensee found that there would be no adverse impact. The MUR power uprate of 1.4 percent is thus bounded by this evaluation since the ventilation heat load increase would be much smaller. Therefore, the containment ventilation system would remain capable of performing its functions following implementation of the proposed 1.4-percent MUR power uprate.

In Point Beach License Amendment Nos. 206 and 211, dated November 26, 2002, the NRC staff accepted the licensee's containment analysis, for which a power level of 102 percent of rated thermal power was assumed. The approved containment analyses bounds the proposed 1.4-percent MUR power uprate.

2.9 Human Performance

2.9.1 Regulatory Evaluation

The human performance evaluation is to ensure that operator performance would not be adversely affected as a result of changes needed for the proposed 1.4-percent MUR power uprate. The following were used to review the proposed action: (1) SRP Chapter 13.2.1, "Reactor Operator Training," (2) SRP Chapter 13.5.2.1, "Operating and Emergency Operating Procedures" (EOPs), and (3) SRP Chapter 18, "Human Factors Engineering."

2.9.2 Technical Evaluation

2.9.2.1 Operator Actions

The licensee indicated that the proposed 1.4-percent MUR power uprate is not expected to have any significant effect on the manner in which the operators control the plant during normal operations or transient conditions, and that all operator actions that were taken credit for in the safety analysis would still be valid following implementation of the proposed 1.4-percent MUR power uprate. Additionally, the licensee evaluated the impact of the proposed 1.4-percent MUR power uprate on the Point Beach Probabilistic Safety Assessment model. The licensee's evaluation also included a review of the Human Reliability Analysis Probabilistic Risk Assessment Notebook. The licensee found that the proposed 1.4-percent MUR power uprate would cause no changes in the timing for operator actions assumed in the Probabilistic Safety Assessment.

2.9.2.2 Emergency and Abnormal Operating Procedures

The licensee indicated that there are currently no EOPs that reference use of the LEFM. Additionally, use of the LEFM would not be incorporated into any EOP. Abnormal Operating Procedure, AOP-21, "PPCS Malfunction," is referred to when either the PPCS or certain PPCS monitoring functions become unavailable. The licensee indicated that a revision to this procedure will direct operators to the proposed TRM Section 3.3.2, which describes the out-of-service actions and describes the TRM Limiting Condition for Operation (TLCO) for an inoperable LEFM. This procedure, as well as normal operating procedures, would be changed to reference the TRM, as appropriate, for LEFM operability.

2.9.2.3 Control Room Controls, Displays and Alarms

The licensee indicated that the installation and use of the LEFM would add a PPCS alarm input for the RTO calculation using the LEFM. This alarm will be audible and will use the existing "Computer Priority Alarm." This alarm functions to alert the operators of PPCS points being out of service as well as a PPCS malfunction. The annunciator position on the control boards would not change. There are no new controls for the operator to manipulate. The Alarm Response Book (ARB) would be updated accordingly. The licensee indicated that reactor operators would be trained on the changes in the PPCS, alarms associated with the LEFM, and the changes in the ARB in a manner consistent with the design modification process.

2.9.2.4 Control Room Plant Reference Simulator

The proposed MUR power uprate is not expected to have a significant effect on any simulated systems. The licensee indicated that changes to the simulator associated with the MUR power uprate would be treated in a manner consistent with any other plant modification, and would be tested and documented accordingly.

2.9.2.5 Operator Training Program

The installation of the LEFM and implementation of the proposed 1.4-percent MUR power uprate would require procedure and training changes. Actions would be added to the appropriate operating procedures and to the TRM in the event the LEFM system becomes unavailable. Operations training concerning the use of the LEFM, the associated procedures and TRM changes, and the increased RTP would be completed prior to implementation of the MUR power uprate. All this information would be updated in a manner consistent with other plant modifications and license amendments.

2.9.3 Conclusion

The NRC staff finds that the licensee has satisfactorily addressed the areas discussed above, and that the proposed 1.4-percent MUR power uprate should not adversely affect human performance. Therefore, the NRC staff finds the proposed 1.4-percent power uprate acceptable with respect to human performance

3.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

The licensee proposed to make conforming changes to Point Beach Facility Operating License Nos. DPR-24 (Unit 1) and DPR-27 (Unit 2) and the TSs to reflect the proposed 1.4-percent power uprate. Below is a summary of the changes.

(1) Revise paragraph 3.A. of the Operating Licenses to authorize operation at reactor core power levels up to, but not in excess of, 1540 MWt.

The change is reflected in page 3 for Unit 1 and page 2 for Unit 2.

Justification for Change:

This change is to reflect the proposed 1.4-percent MUR power uprate, which corresponds to a maximum licensed power level of 1540 MWt.

(2) Revise TS 1.1, "Definitions," for "RATED THERMAL POWER (RTP)" to reflect the increase from 1518.5 MWt to 1540 MWt.

The change is reflected on TS page 1.1-5.

Justification for Change:

This change is to reflect the proposed 1.4-percent MUR power uprate, which corresponds to a maximum licensed power level of 1540 MWt.

(3) Revise TS 5.6.4, "CORE OPERATING LIMITS REPORT (COLR)," as follows:

Add references (11) and (12) to TS 5.6.4.b for Caldon Topical Reports ER-80P and ER-160P, respectively.

Revise reference (4) of TS 5.6.4.b, WCAP-14787-P, Revision 0, Revised Thermal Design Procedure, to refer to Revision 2. Revision 2 contains the LEFM's power measurement uncertainty.

Revise the text of TS 5.6.4.b to explain the use of the LEFM power measurement uncertainty in other topical reports listed in the COLR. As stated in Section 1.1, "Background," of the April 30, 2002, application, the licensee proposes continued use of the topical reports identified in this TS. These reports describe NRC-approved methods that support the Point Beach safety analyses. In some of these topical reports, such as the SBLOCA and LBLOCA reports, reference is made to the use of the 2-percent power uncertainty that is consistent with the original Appendix K rule. The licensee proposes these topical reports be approved for use consistent with the new Appendix K rule and the proposed 1.4-percent MUR power uprate (i.e., using 0.6 percent uncertainty instead of the 2-percent power measurement uncertainty). To describe this change in applying the power measurement uncertainty, the following text will be inserted just prior to the listing of topical reports in TS 5.6.4.b:

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of the original rated thermal power is specified in a previously approved method, 100.6 percent of uprated rated thermal power may be used only when the main feedwater flow measurement (used as the input for reactor thermal output) is provided by the Caldon leading edge flowmeter (LEFM) as described in reports 11 and 12 listed below. When main feedwater flow measurements from the LEFM are unavailable, a power measurement uncertainty consistent with the instruments used shall be applied.

Future revisions of approved analytical methods listed in this Technical Specification that currently reference the original Appendix K uncertainty of 102 percent of the original rated thermal power should include the condition given above allowing use of 100.6 percent of uprated rated thermal power in the safety analysis methodology when the LEFM is used for main feedwater flow measurement.

The approved analytical methods are described in the following documents:

The changes are reflected on TS pages 5.6.3, 5.6.4, 5.6.5, and 5.6.6.

Justification for Change:

This change is to reflect the proposed 1.4-percent MUR power uprate, which corresponds to a maximum licensed power level of 1540 MWt.

Conclusion

The NRC staff's evaluation, as reflected in this safety evaluation, supports the licensee's proposed changes to the Operating Licenses and TSs and the NRC staff, therefore, finds the proposed changes acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational

radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (67 FR 57630). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 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.

7.0 REFERENCES

1. Caldon Topical Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM™ System," March 1997.

2. Caldon Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM™ or CheckPlus™ System," Revision 2, December 2000.

3. Westinghouse Licensing Topical Report WCAP-11394 (Proprietary) and WCAP-11395 (Nonproprietary), "Methodology for the Analysis of the Dropped Rod Event," October 23, 1989.

4. Westinghouse Licensing Topical Report WCAP-10054-P-A (Proprietary) and WCAP-10081-NP (Nonproprietary), Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997.

5. Westinghouse Licensing Topical Report WCAP-14449-P, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," October 29, 1999.

6. Letter from G. Hatchett, NRC, to M. Sellman, Wisconsin Electric Power Company, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments [193 (Unit 1) and 198 (Unit 2)] RE: Design and Operation of Fuel Cycles with Upgraded Westinghouse Fuel (TAC Nos. MA5939 AND MA5940)," February 8, 2000.

7. Westinghouse Letter NS-TMA-2182, “Anticipated Transients Without Scram for Westinghouse Plants,” December 1979.

8. Caldon Topical Report ER-160P, "Supplement to Topical Report ER-80P: Basis for Power Uprate with the LEFM ✓™ System," May 2000.

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