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Committed to Nuclear Excellence

Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

NRC 2002-0030

10 CFR 50.90

April 30, 2002

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Ladies/Gentlemen:

Dockets 50-266 And 50-301
License Amendment Request 226
Measurement Uncertainty Recapture Power Uprate
Point Beach Nuclear Plant, Units 1 And 2

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) requests amendments to the operating licenses and the plant Technical Specifications for Point Beach Nuclear Plant (PBNP) Units 1 and 2 to incorporate an increase in licensed reactor thermal power (RTP) level. The requested increase in licensed RTP is the result of a measurement uncertainty recapture (MUR) power uprate. The information provided in support of this request is based on NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The proposed change would increase the licensed reactor core power level by 1.4 percent from 1518.5 MWt to 1540 MWt. The NMC's request is based on reduced uncertainty in the reactor thermal output (RTO) measurement achieved by installation of a Caldon, Incorporated LEFM✓™ 2000FC Flow Measurement System (LEFM) in the main feedwater systems of both PBNP units. The reduced power measurement uncertainty allows for a power uprate that is equivalent to the 10 CFR 50 Appendix K criteria of two percent minus the calculated LEFM-based power measurement uncertainty of 0.6 percent. Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," documents the theory, design, and operating features of the LEFM system and its ability to achieve increased accuracy in main feedwater flow measurement. Topical Report ER-80P is supplemented by ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ System." The NRC approved ER-80P for referencing in power uprate applications in a safety evaluation dated March 8, 1999, for Comanche Peak. On January 19, 2001, the NRC also approved the use of ER-160P in the safety evaluation for the Watts Bar MUR power uprate application. Several other licensees have referenced the Caldon Topical Reports in MUR uprate applications since the issuance of the above mentioned NRC safety evaluations.

A001

Rec'd
6/18/02

This amendment request is supported by several enclosed attachments. Attachment 1 provides a description and assessment of the proposed amendment and is based on the NRC MUR application guidance. The content of Attachment 1 is structured to facilitate easier review by the NRC and is broken down as follows:

Section 1.0 contains the Introduction and Background Information.
Section 2.0 describes the License and Technical Specification Changes.
Section 3.0 involves the Technical Assessment of the Change in RTP (based on the NRC guidance of RIS 2002-03) and an Environmental Review.
Section 4.0 contains a No Significant Hazards Determination.

Additionally, Westinghouse Electric Company has performed the calorimetric power measurement uncertainty calculations for use with the LEFM. Proprietary and Non-Proprietary summaries of the uncertainty calculations are provided in Attachments 2 (WCAP-14787, Revision 1) and 3 (WCAP-14788, Revision 1), respectively. Both reports in Attachments 2 and 3 are titled, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology For Wisconsin Electric Power Company Point Peach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 Mwt-NSSS Power with Feedwater Venturis, or 1679 Mwt-NSSS Power with LEFM on Feedwater Header). Attachment 4 provides the existing Facility Operating License and TS pages marked up to show the proposed change. Attachment 5 provides revised (clean) Facility Operating License and TS pages. There are no TS bases changes. Attachment 6 contains a list of regulatory commitments associated with this proposed amendment.

Also enclosed are a Westinghouse authorization letter, CAW-02-1516 (Attachment 7), an accompanying affidavit (Attachment 8), Proprietary Information Notice (Attachment 9), and Copyright Notice (Attachment 10).

As Attachment 2 contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of 10 CFR 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or supporting the Westinghouse Affidavit, should reference the appropriate authorization letter and be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

NMC requests approval of this proposed amendment by October 31, 2002. Upon NRC approval of this proposed change, NMC requests that the amendment be made effective on the date of issuance, but allow an implementation period of ninety days to provide sufficient time for associated administrative activities. This would allow implementation of the uprate for both PBNP Units in the last quarter of 2002. The approval date was selected based on LEFM installation planned by August 2002 and the Unit 1 refueling outage, scheduled to end October 2002. A late October approval would allow the uprate to be implemented on both units shortly after the Unit 1 start up. This date still allows a reasonable time for NRC review and allows PBNP to take advantage of the economic benefits of the uprate as soon as possible. It should be noted that the plant does not require this amendment to allow continued safe, full power operation.

The NMC has determined that the information for the proposed amendments does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact appraisal need not be prepared.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Wisconsin Official.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects, these statements are not based entirely on my personal knowledge, but on information furnished by cognizant NMC employees and consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on April 30, 2002.

Mark E Warner

Mark E. Warner
Site Vice President

LMG

Attachments:
1 - Description and Assessment
2 - WCAP–14787, Rev. 1, Proprietary
3 - WCAP-14788. Rev. 1, Non-Proprietary
4 - Proposed Facility Operating License and Technical Specification Changes (marked up)
5 - Proposed Facility Operating License and Technical Specification Changes (clean copies)
6 - List of Regulatory Commitments
7 - Westinghouse Proprietary Authorization Letter, CAW-02-1516
8 - Westinghouse Affidavit
9 - Westinghouse Proprietary Information Notice
10 - Westinghouse Copyright Notice

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

bcc:

R. A. Anderson	A. J. Cayia	K. M. Duescher (3)
R. R. Grigg (P460)	L. Armstrong	L. Schofield (JOSRC)
K. E. Peveler	R. P. Pulec	M. E. Reddemann
D. A. Weaver (P129)	T. J. Webb	E. J. Weinkam III
T. J. Taylor	M. E. Warner	File
L. Gunderson	H. Hanneman	

ATTACHMENT 1

To

Letter from Mark Warner (NMC)

to

Document Control Desk (NRC)

<u>License Amendment Request 226</u>

Description and Assessment

1.0 INTRODUCTION AND BACKGROUND INFORMATION

The Point Beach Nuclear Plant (PBNP) Units 1 and 2 are currently licensed to operate at a maximum rated thermal power (RTP) of 1518.5 MWt. Approval is being requested to increase the licensed RTP by 1.4 percent to 1540 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the reactor thermal output (RTO) of each reactor unit. Increasing RTP by reducing measurement uncertainty is called a measurement uncertainty recapture (MUR) power uprate. The Nuclear Management Company, LLC (NMC), has evaluated the impact of a 1.4 percent uprate to 1540 MWt for the applicable systems, structures, components, and safety analyses at PBNP. The results of this evaluation and the new main feedwater flow measurement system are described in the following assessment in this attachment (Attachment 1, "Description and Assessment").

1.1 Background

The 1.4 percent power uprate for PBNP is based on eliminating unnecessary analytical margin that is assumed in analyses to account for the measurement uncertainties associated with measuring the RTO of each unit. Point Beach's current accident and transient analyses include a minimum two percent margin on RTP to account for power measurement uncertainty. This power measurement uncertainty was originally required by 10 CFR 50, Appendix K, "ECCS Evaluation Models." The rule required a two percent power margin between the licensed power level and the power level assumed for the ECCS evaluations. In 2000, the NRC amended Appendix K to provide licensees the option of maintaining the two percent power margin or applying a reduced margin. For the latter case, the new assumed power level had to account for measurement uncertainties in the power level measurement instrumentation. The revised Appendix K rule had an effective date of July 31, 2000. Uprates taking advantage of this rule change are referred to as MUR uprates, calorimetric uprates, or mini-uprates.

Uncertainty in the main feedwater flow measurement is the most significant contributor to RTO power measurement uncertainty. Based on this fact and on the above Appendix K rule change, the PBNP proposes a reduced power measurement uncertainty of 0.6 percent and an increase in RTP of 1.4 percent. To accomplish this reduction in uncertainty and increase in power, the PBNP will install a Caldon, Incorporated (Caldon) LEFM✓™ 2000FC Flow Measurement System (LEFM) for measuring the main feedwater flow in PBNP Units 1 and 2. The Caldon LEFM provides a more accurate measurement of feedwater flow than that assumed during the development of the original Appendix K requirements and that of the feedwater flow venturis currently used to calculate RTO. The LEFM will measure feedwater mass flow to within ±0.43 percent for Unit 1 and ±0.41 percent for Unit 2. However, a bounding feedwater mass flow uncertainty of ±0.48 was used to calculate a total power measurement uncertainty of ±0.58 percent. Based on this, PBNP proposes to reduce the power measurement uncertainty required by Appendix K to 0.6 percent. This value (i.e., 0.6 percent) has been used in the PBNP safety analyses supporting this license amendment request. The improved power measurement uncertainty obviates the need for the two percent power margin originally required by Appendix K, thereby allowing an increase in the RTP available for electrical generation by 1.4 percent.

Two Caldon Topical Reports technically support performing a power uprate by improving the accuracy of instrumentation used in measuring RTO. Caldon Engineering Report 80P (ER-80P), "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," supports Appendix K uprates of up to one percent. This Caldon report was approved in the NRC Safety Evaluation Report (SER) dated March 8, 1999, for support of TU Electric's (Comanche Peak) exemption request from the original Appendix K requirements and for a one percent MUR uprate. Caldon ER-80P is supplemented by Caldon Engineering Report 160P (ER-160P), "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System." The supplement ER-160P was published subsequent to the Appendix K rule change and supports power increases of up to 1.4 percent using the Caldon LEFM✓™ system in measuring RTO. This report was approved in the NRC SER for the Watts Bar MUR power uprate dated January 19, 2001.

In addition to the proposal to increase the RTP to 1540 MWt, the NMC also proposes continued use of the topical reports identified in the PBNP Units 1 and 2 Technical Specification (TS) 5.6.4.b and the PBNP Core Operating Limits Reports. The reports referenced in this TS and in the COLR describe the NRC approved analytical methodologies used to determine the core operating limits for PBNP Units 1 and 2. This includes the small and large break loss of coolant accidents. In some of these topical reports, reference is made to the use of a two percent power measurement uncertainty being applied consistent with 10 CFR 50, Appendix K. The NMC requests that these topical reports be approved for use consistent with this license amendment (i.e., 0.6 percent power measurement uncertainty be assumed instead of 2 percent). The proposed change is further described in Section 2.0 of this attachment. Additionally, the reduction of the power measurement uncertainty does not constitute a significant change as defined in 10 CFR 50.46(a)(3)(i) regarding emergency core cooling system (ECCS) evaluation models.

1.2 Licensing Methodologies for Uprate

The analytical and licensing work supporting the PBNP MUR power uprate is consistent with the methodology established by Westinghouse in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," dated 1983. The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects including the broad categories that must be addressed. These categories include the Nuclear Steam Supply System (NSSS) parameters, systems, components, design transients, accidents, fuel, and the interfaces between the NSSS and the Balance of Plant (BOP) systems. This methodology provides key points that promote correctness, consistency, and licensability in a power uprate program. The key points include the use of well-defined analysis input assumptions and parameter values, the use of currently approved analytical techniques, and the use of currently applicable licensing criteria and standards. This methodology has been successfully used as the basis for power uprate projects on over twenty pressurized water reactors, including measurement uncertainty recapture uprates.

The proposed PBNP MUR power uprate is also consistent with Caldon, Incorporated Engineering Reports ER-80P and ER-160P. The NRC has approved both of these topical reports. Point Beach is specifically applying these Topical Reports, and the criteria listed in the NRC SER for the Caldon Topical Report ER-80P, for a requested 1.4 percent RTP increase.

In addition to the above methodologies, PBNP has taken into account the specific guidance developed by the NRC for the content of MUR power uprate applications. This guidance was published on January 31, 2002, as NRC RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." Therefore, this license amendment application is structured to be consistent with the NRC guidance.

1.3 Licensing Approach to Plant Safety, Component, and System Analyses

The reactor core power and the NSSS thermal power are used as inputs to most plant safety, component, and system analyses. Generally, the PBNP analyses model the core and the NSSS thermal power in one of three ways:

1) Some of the analyses apply a two percent uncertainty to the initial licensed power level of 1518.5 MWt to account solely for the power measurement uncertainty. This results in an assumed core power level of 1549 MWt in the analyses. These analyses have not been reperformed for the 1.4 percent uprate since the sum of the requested core power level (1540 MWt) plus the decreased power measurement uncertainty (0.6 percent) result in an assumed core power level of 1549 MWt. Therefore, these analyses fall within the previously analyzed conditions.

2) Some of the analyses already employ an assumed core power level in excess of the requested 1540 MWt plus the new power measurement uncertainty of 0.6 percent (i.e., 1549 MWt). These analyses were performed at 1549 MWt core power or greater (with or without the two percent measurement uncertainty) during previous plant programs (i.e., steam generator replacement, fuel upgrade, etc.) reviewed and approved by the NRC. The higher power level was assumed in these programs to account for potential future power uprates and plant changes. For these analyses, the available margin envelopes the 1.4 percent power increase of the MUR uprate. Consequently, these analyses have not been reperformed and continue to retain sufficient analysis margin.

3) The remaining analyses are performed at zero percent power conditions or do not actually model the core power level. These analyses have not been reperformed since they are unaffected by the core power level.

1.4 Conclusion

NMC is requesting a 1.4 percent increase in RTP for PBNP Units 1 and 2 from 1518.5 MWt to 1540 MWt. This power increase will be accomplished by using a more accurate main feedwater flow measurement system to calculate the RTO of each reactor unit. This higher accuracy measurement will be achieved with the use of a Caldon LEFM measurement system. This license amendment request has taken into account industry and NRC accepted methodologies and guidelines for power uprates.

This License Amendment Request (LAR) is made pursuant to 10 CFR 50.90 to modify the Operating Licenses and the TS requirements associated with rated thermal power and the use of the power measurement uncertainty in safety analyses.

2.0 DESCRIPTION OF LICENSE AND TECHNICAL SPECIFICATION CHANGES

The proposed license amendment will revise the PBNP Unit 1 and 2 Facility Operating Licenses and the Technical Specifications (TS) to increase the licensed RTP by 1.4 percent from 1518.5 MWt to 1540 MWt. The proposed changes are described in detail below and indicated on the marked up and clean copy Operating License and TS pages in Attachments 4 and 5.

2.1 Revise paragraph 3.A. of the operating licenses for both units, 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.

2.2 Revise TS 1.1, stating the definition of RATED THERMAL POWER (RTP), to reflect the increase from 1518.5 MWt to 1540 MWt.

2.3 Revise TS 5.6.4, Core Operating Limits Report (COLR) as follows:

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

b) Revise reference (4) of TS 5.6.4.b, WCAP-14787-P, Revision 0, Revised Thermal Design Procedure, to refer to Revision 1. Revision 1 contains the LEFM's power measurement uncertainty.

c) 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," PBNP proposes continued use of the topical reports identified in this TS. These reports describe NRC approved methods that support the PBNP safety analyses. In some of these topical reports, such as the small and large break loss of coolant accident reports, reference is made to the use of the two percent power uncertainty that is consistent with the original Appendix K rule. PBNP proposes these topical reports be approved for use consistent with the new Appendix K rule and this amendment request (i.e., using 0.6 percent uncertainty instead of the two 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:"

3.0 TECHNICAL ASSESSMENT OF THE CHANGE IN RTP

The technical assessment is broken out as designated by the NRC submittal guidance in RIS 2002-003.

Section	Subject
3.1	Feedwater Flow Measurement Technique and Power Measurement Uncertainty
3.2	Accidents and Transients that Bound Plant Operation at the Proposed Uprated Power Level
3.3	Accidents and Transients Not Bounded by Current Analyses
3.4	Mechanical/Structural/Material Component Integrity and Design
3.5	Electrical Equipment Design
3.6	System Design
3.7	Other Evaluations
3.8	Technical Specification, Protection System Setting, and Emergency System Setting Changes

3.1 Feedwater Flow Measurement Technique and Power Measurement Uncertainty

3.1.1 Identification of Approved Topical Reports and NRC SERs

The use of a Caldon LEFM✓™ 2000FC Flow Measurement System (LEFM) in determining and monitoring main feedwater flow in nuclear power plants has been approved by the NRC for MUR power uprate applications. The approval is documented in the March 8, 1999, Comanche Peak SER for Caldon Topical Report ER-80P, Rev 0, March 1997, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System." The information in ER-80P was further supplemented by topical report ER-160P, Revision 0, May 2000, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM System." The ER-160P was approved in the Watts Bar MUR Uprate SER dated January 19, 2001.

3.1.2 Description of the Caldon Leading Edge Flow Meter (LEFM), the LEFM✓™ 2000FC Flow Measurement System

The Caldon LEFM✓™ (LEFM) is an ultrasonic flow meter that determines fluid flow by measuring the transit time of ultrasonic pulses as they travel through the flowing fluid. The LEFM consists of a spool piece for each unit and an electronics processing unit. The spool piece contains eight ultrasonic transducer assemblies that are mounted in separate transducer housings to form four parallel chordal acoustical paths across the spool piece. Each transducer may be removed at full power conditions without disturbing the pressure boundary. These transducers are mounted in the LEFM spool piece at an angle of 45 degrees to the flow. The ultrasonic transducers send and receive acoustic energy pulses. The pulses traverse the pipe faster with the flow and slower against the flow. The LEFM then uses the transient times and the time differences between the pulses to determine the fluid velocity and temperature.

At PBNP Units 1 and 2, a spool piece is permanently installed in the common 20 inch Main Feedwater header to monitor total feed flow for each unit. These spool pieces were installed in the early 1980s and have been used for several different LEFM systems. Although the current transducers in the spool piece could be used with the new Caldon LEFM✓™ 2000FC Flow Measurement System, all are being upgraded to a more reliable transducer assembly design.

The digital electronics processing unit performs the following functions:

1) sends, receives, and detects ultrasonic pulses;
2) measures the pulse's transit time;
3) performs the flow and temperature calculations from the transit time data;
4) provides local display and control;
5) performs checks for on-line verification; and
6) contains the necessary output devices to provide the operator with flow and temperature data and on-line verification through the means of display screens and alarms.

The software controlling the unit employs the ultrasonic transit time method to measure four line integral velocities at precise locations with respect to the pipe centerline. The system numerically integrates the four velocities measured according to the method described in Caldon's Topical Report ER-80P.

One LEFM electronics unit, which contains an individual CPU (central processing unit) for each PBNP unit, supports both spool pieces. The electronics unit will take data from each spool piece and process it in the corresponding CPU. The main feedwater mass flow rate will be calculated and displayed on the local display panel. The data will also be transmitted to the Plant Process Computer System (PPCS). The main feedwater mass flow rate, main feedwater temperature, and verification status will be used by the PPCS to determine the RTO, to calculate the feedwater venturi correction factor, and for PPCS and control room alarm functions.

3.1.3 Plant Specific Use of the Caldon LEFM to Determine Calorimetric Power

The LEFM is an improved system for determining and monitoring feedwater flow in nuclear power plants. The technology, detailed in ER-80P and ER-160P, provides significantly higher accuracy and reliability than flow instruments that use differential pressure measurements and resistance temperature detector (RTD) instruments.

The LEFM provides measurements of feedwater mass flow to within ±0.43 percent for Unit 1 and ±0.41 percent for Unit 2. These values specifically apply to the PBNP LEFM. The Caldon report, ER-80P, assumes a bounding feedwater mass flow uncertainty of ±0.48 percent. The ER-80P feedwater mass flow uncertainty bounds the PBNP specific uncertainty values. The bounding feedwater mass flow uncertainty of ER-80P supports a total power measurement uncertainty of ±0.58 percent as calculated by Westinghouse for the Revised Thermal Design Procedure (Attachments 2 and 3). This power measurement uncertainty is substantially more accurate than the ±2 percent power measurement uncertainty typically applied to the conventional venturi-based instrumentation. It is also conservative since the generic bounding feedwater mass flow uncertainty was used in the power measurement uncertainty calculation rather than the site-specific uncertainties. The smaller power measurement uncertainty of ±0.58 percent (rounded conservatively to 0.6 percent) allows for an increase in RTP by 1.4 percent, or the difference between the currently assumed venturi-based power measurement uncertainty and that of the LEFM.

The LEFM measurements of main feedwater mass flow and main feedwater temperature will be directly substituted for the venturi-based mass flow measurement and the RTD temperature measurement currently used in the plant secondary side calorimetric heat balance calculation. These parameters will be directly input to the PPCS and used by the PPCS software to calculate the RTO. This power determination will be used to adjust the Nuclear Instrumentation System (NIS) channels daily (every 24 hours) in accordance with the Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.2 and to ensure power operation at or below RTP. The venturi-based feedwater flow measurement will continue to be used for feedwater control, reactor protection, and other functions that it currently fulfills. Additionally, the venturi-based main feedwater flow will be corrected on the basis of the LEFM feedwater flow measurement. The correction factor will be based on a correction factor average developed by comparison of the LEFM feedwater flow measurement and the venturi feedwater flow measurement. This will allow the venturis to serve as an accurate feedwater flow input to the RTO determination in the event that the LEFM system is not available.

Additionally, the LEFM provides on-line verification of the accuracy of the main feedwater flow and temperature measurements upon which NSSS thermal power determinations are based. This on-line verification feature provides assurance that LEFM performance is consistent with the design basis.

3.1.4 NRC SER Criteria for Caldon LEFM

The next four sections specifically show compliance with the NRC SER for the Caldon LEFM system. Additionally, the RIS guidance was combined with these criteria where appropriate.

Maintenance and Calibration (ER-80P SER Criterion 1/NRC RIS 2002-03, I.1.F)

Calibration, Maintenance, Configuration Control and Corrective Actions

Calibration and maintenance of the LEFM system will be performed using site procedures developed from the requirements of the Caldon Technical Manual. Incorporation of, and adherence to, these requirements will assure that the LEFM is properly maintained and calibrated. These changes will be made prior to implementing the MUR power uprate. All equipment problems associated with the LEFM will be controlled under the site work order process, or the corrective action program. Verification of system operation is provided by system diagnostics. All hardware modifications for the LEFM or PPCS will be performed in accordance with the site engineering change or modification processes.

Other instrumentation used in the calorimetric power uncertainty calculation are calibrated every refueling outage through the use of appropriate instrument and control procedures. Problems encountered during the calibration procedures will be handled by the appropriate plant process.

Software Control and Deficiency Reporting

Although the LEFM's calorimetric input is not nuclear safety-related, the system's software has been developed and is maintained under Caldon's 10 CFR 50 Appendix B Quality Assurance Program and its Verification and Validation (V&V) program. The Caldon V&V program is consistent with IEEE 7-4.3.2 (1993), Annex E, and ASME NQA-1-1999, Addenda Subpart 2.7. The V&V program is also consistent with guidance for software V&V in EPRI TR-103291s, "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. Additionally, Caldon's V&V Program and Quality Program procedure address notification of important defects or inconsistencies in Caldon's software that could affect the design basis accuracy of the LEFM. The corrective action program at PBNP will address any deficiency reports of concern received from Caldon. Caldon's V&V program is briefly described in Topical Report ER-80P, Section 6.4, "Quality Measures in Design, Fabrication, and Factory Acceptance Testing of the LEFM."

Point Beach Nuclear Plant's software control for the LEFM and the PPCS will be in accordance with the site software management program. This program includes guidance for reporting software deficiencies to vendors.

Operational and Maintenance History (ER-80P SER Criterion 2)

The Point Beach Nuclear Plant has had ultrasonic feedwater flow measurement systems installed since the early 1980s. However, the systems have been older models not representative of the LEFM system that will be installed and relied upon for the MUR power uprate. Therefore, plant specific maintenance and operations data is not applicable to the new system. However, significant operational experience has been accumulated from several nuclear power plant applications. The cumulative operating history shows the Caldon LEFM system reliability. This supporting information is summarized in Section 7.0 of ER-80P. The Caldon LEFM system that will be installed at PBNP Units 1 and 2 is representative of the LEFM Check system described in Topical Reports ER-80P and ER-160P and is bounded by the requirements set forth in those reports.

Uncertainty Determination Methodology (ER-80P SER Criterion 3/NRC RIS 2002-03, I.1.E)

Caldon, Incorporated provided main feedwater mass flow and main feedwater temperature uncertainties to Westinghouse Electric Company for performance of the power measurement uncertainty calculations. The feedwater flow and temperature measurement uncertainties were taken from the Caldon Engineering Report ER-80P as the bounding uncertainty for an LEFM installed on a common feedwater header. The uncertainties used bounding values that are conservative. The bounding feedwater mass flow uncertainty provided by Caldon Engineering Report ER-80P was ±0.48 percent. Later, Caldon specifically calculated the feedwater mass flow uncertainties for PBNP. The PBNP specific feedwater mass flow uncertainties are ±0.43 percent for Unit 1 and ±0.41 percent for Unit 2. These values are bounded by the value given to Westinghouse Electric Company to perform the power measurement uncertainty calculations.

Westinghouse Electric Company completed calculations of the power measurement uncertainty for PBNP Units 1 and 2. Westinghouse calculated the power uncertainties using the Westinghouse Revised Thermal Design Procedure (RTDP) instrument uncertainty methodology. The RTDP methodology was previously performed for the PBNP using the venturi measured feedwater flow for the power measurement uncertainty. The WCAP associated with the previous RTDP uncertainty work was submitted to the NRC during the Fuel Upgrade Project and was subsequently accepted by the NRC on 02/08/2000. Therefore, the RTDP is a currently accepted plant setpoint methodology.

The original RTDP WCAP has been revised to support uprated power levels. Revision 1 of the WCAP includes the power uncertainty analysis using both the venturi and the new LEFM. The report also identifies that the uncertainties calculated are applicable for power levels up to 1673 MWt when the daily calorimetric power measurement is based on the LEFM feedwater flow measurement. The report identifies all power measurement parameters and their individual contribution to the power measurement uncertainty. Proprietary and non-proprietary versions of this report are attached as Attachments 2 and 3 (WCAP-14787, Revision 1, and WCAP-14788, Revision 1, both titled, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 and 2 (Fuel Upgrade and Uprate to 1656 MWt-NSSS Power with the Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header)").

Hydraulic Modeling/Use of Site Specific Piping Configuration (ER-80P SER Criterion 4)

The Point Beach LEFM spool pieces were installed in the early 1980s by Westinghouse and were not calibrated in a site-specific hydraulic geometry prior to installation. Therefore, the PBNP spool piece profile factors were established using a statistical approach. The approach is described in detail in Caldon report ER-80P that was approved by the NRC. Further, a site specific detailed analysis of actual profile measurements has been made and documented in Appendix B of MPR Associates Report, MPR-1619, "Feedwater Flow Measurement with LEFM Chordal Systems at Point Beach Units 1 and 2, Configuration and Uncertainty Analysis," dated May 1995. The conclusion of MPR-1619 is that the uncertainty of the Hydraulic Profile Factor for the Point Beach LEFM Systems is ±0.32 percent. This value is less than the ±0.4 percent bounding value in ER-80P.

3.1.5 LEFM Failure, Proposed Outage Time, and Proposed TRM Actions

LEFM system failures are detected and transmitted to the PPCS and will cause an audible alarm in the control room. Additionally, a control board annunciator will be lit for the priority computer alarm. This annunciator alarms in the event the LEFM fails, the PPCS malfunctions, or the PPCS monitoring functions become unavailable. The LEFM system does not perform any safety function and is not used to directly control any plant systems. Therefore, system inoperability has no immediate effect on plant operation.

PBNP will be operated in accordance with the safety analyses and the applicable power measurement uncertainty. When the LEFM is available, the LEFM-based calorimetric uncertainty of 0.6 percent will be used and the plant will be operated at or below an RTP of 1540 MWt. Based on the LEFM uncertainty of 0.6 percent, initial power assumed in certain accident analyses would be 1549 MWt. Additionally, the PPCS software will calculate a correction factor for the venturi feedwater flow that will result in the venturi flow equaling the LEFM flow.

If the LEFM system becomes unavailable, the operators will receive the priority computer alarm and be directed to the Alarm Response Book (ARB). The ARB will direct operators to the Technical Requirements Manual (TRM) Section 3.2.2 for an inoperable LEFM. The TRM will have an action statement that requires the LEFM to be returned to service prior to the next performance of TS SR 3.3.1.2 (NIS channel surveillance). The basis of the time period is the TS SR 3.3.1.2 completion time (24 hours plus 25 percent). The NIS will have been adjusted with the last LEFM-based RTO calculation and will be acceptable until the next performance of the surveillance. Operations will be using the LEFM-corrected venturi feedwater flow and temperature for the RTO calculation. Plant operations may remain at an RTO of 1540 MWt while continuing to use the LEFM corrected venturi RTO calculation prior to the next performance of TS SR 3.3.1.2. To remain in compliance with the basis for the operation with the RTP of 1540 MWt, the LEFM must be returned to service prior to the next performance of TS SR 3.3.1.2. If the LEFM is not returned to service within the above time, the TRM would require power be reduced or maintained at the power level associated with the use of the feedwater venturis. This power level is 1520 MWt and is consistent with the calculated power measurement uncertainty associated with the feedwater venturis (as calculated by WCAP-14787). The NIS surveillance would then be performed using the venturi-based feedwater flow measurements. Once TS SR 3.3.1.2 is performed using the corrected venturi-based feedwater flow measurement, the assumed power measurement uncertainty is 1.87 percent. It would be necessary to operate the plant at 1520 MWt until the LEFM is restored. Once the LEFM is returned to service, performance of TS SR 3.3.1.2 using the LEFM RTO calculation would be required before power can be escalated to the 1540 MWt.

The basis for reducing power to 1520 MWt rather than the original RTP of 1518.5 MWt is the relaxation of the Appendix K rule. The change in the rule allows PBNP to use the feedwater venturis calculated power measurement uncertainty of 1.87 percent as opposed to the 2 percent power measurement uncertainty required by the original Appendix K rule. Applying 1.87 percent power measurement uncertainty allows for a 0.13 percent uprate from the current RTP of 1518.5 MWt to 1520 MWt. This power level with the 1.87 percent power measurement uncertainty results in the same initial power level of 1549 MWt for the accident analyses.

Point Beach operations procedures will be revised to reflect the above responses to the unavailability of the LEFM for FW flow measurement. Additionally, this information will be included in operator training prior to implementation of the MUR uprate license amendment.

3.2 Accidents and Transients that Bound Plant Operation at the Proposed Uprated Power Level of 1540 MWt

The PBNP Accident Analyses (FSAR Chapter 14 events) and other licensing-basis analyses were reviewed to determine the effect of the 1.4 percent MUR power uprate. The tables below explicitly state the following: the power level the analysis is performed at, the power uncertainty established for the analysis, the final power level assumed in the analysis, and reference to the applicable NRC approval or appropriate reference.

3.2.1 Bounding Non-LOCA Accident Analyses with No Radiological Consequences

The analyses referenced in this table are the existing licensing basis analyses of record for the plant. None of these analyses are changing and all bounding event determinations continue to remain valid for the MUR Power uprate.

Table 3.2.1-1 Bounding Non-LOCA Accident Analyses

Non-LOCA Accident Analysis (FSAR Section 14.1)		**Current FSAR Power Assumption**				**Is uprate to 1540 MWt RTP bounded? Reference?**
		% Pwr	**Core MWt**	**Uncer-tainty**	**Power Level**	
14.1.1	Uncontrolled RCCA[1] Withdrawal from Subcritical	0	1650	None	0	Yes, Ref. 3.2.5.1
14.1.2	Uncontrolled RCCA[1] Withdrawal at Power	100 60 10	1650	RTDP[2]	1650 990 165	Yes, Ref. 3.2.5.1
14.1.3	RCCA[1] Drop	100	1650	RTDP[2]	1650	Yes, Ref. 3.2.5.1
14.1.4	Chemical and Volume Control System Malfunction (Boron Dilution)	100 5 0	1650	N/A	1650 82.5 0	Yes, Ref. 3.2.5.1
14.1.5	Startup of an Inactive Reactor Coolant Loop	10	1518.5	2%	182.2	Yes, Ref. 3.2.5.1, 3.2.5.14, 3.2.5.12
14.1.6	Reduction in Feedwater Enthalpy Incident	100	1650	RTDP[2]	1650	Yes, Ref. 3.2.5.1
14.1.7	Excessive Load Increase Incident	100	1650	RTDP[2]	1650	Yes, Ref. 3.2.5.1
14.1.8	Loss of Reactor Coolant Flow	100	1650	RTDP[2]	1650	Yes, Ref. 3.2.5.1
	Locked Rotor (core response)	100	1650	2%	1683	Yes, Ref. 3.2.5.1
14.1.9	Loss of External Electrical Load (Overpressure)	100	1650	2%	1683	Yes, Ref. 3.2.5.1
	Loss of External Electrical Load (DNB)	100	1650	RTDP[2]	1650	Yes, Ref. 3.2.5.1
14.1.10	Loss of Normal Feedwater	100	1518.5	2%	1549	Yes, Ref. 3.2.5.1
14.1.11	Loss of All A/C Power to the Station Auxiliaries	100	1650	2%	1683	Yes, Ref. 3.2.5.1
14.1.12	Likelihood of Turbine-Generator Unit Overspeed	100	1650	0%	1673	Yes, Ref. 3.2.5.15

(1) Rod Cluster Control Assembly.
(2) Revised Thermal Design Procedure Methodology used to incorporate power measurement uncertainty (0.6% for LEFM-based and 2.0% for venturi-based).

Analyses in Table 3.2.1-1 above that reference RTDP (Revised Thermal Design Procedure) power uncertainty use the RTDP methodology to predict the plant's DNBR design limit. This is an NRC approved method of modeling uncertainties on pressurizer pressure, primary coolant temperature, reactor coolant system flow, and reactor power. As stated in section 3.1.5, the RTDP was used for calculating power measurement uncertainty using both the LEFM and the venturis. It is a currently accepted methodology for PBNP.

Additionally, FSAR 14.1.5, "Startup of an Inactive Reactor Coolant Loop," is based on a historical analysis because the PBNP TS 3.4.4, "RCS Loops – Modes 1 and 2," requires two RCS loops shall be operable and in operation in these modes. Therefore, the analysis did not have to be reperformed for the MUR power uprate.

3.2.2 Bounding Non-LOCA Accidents with Potential Radiological Consequences

The review of non-LOCA Chapter 14 analyses with radiological consequences shows that all analyses of record remain bounding. All were performed at a core power of 1650 MWt with the exception of the Fuel Handling Accident (FHA). The FHA was performed at 1518.5 MWt and includes a 2 percent uncertainty. None of these analyses are changing and the bounding event radiological consequences continue to remain valid for the MUR power uprate.

Table 3.2.2-1 Non-LOCA Accident Analyses with Radiological Consequences Bounded by Current Analyses

Non-LOCA Analyses with Potential Radiological Consequences (FSAR Sections 14.1 and 14.2)		**Current FSAR Power Assumption**				**Is uprate to 1540 MWt RTP bounded? Reference?**
		% Pwr	**Core MWt**	**Uncer-tainty**	**Power Level**	
14.2.1	Fuel Handling Accident	Shut down (SD)	1518.5 prior to SD	2%	1549	Yes, Ref. 3.2.5.1
14.2.2	Accidental Release-Recycle or Waste Liquid	N/A	N/A	N/A	N/A	Yes. Ref. 3.2.5.11
14.2.4	Steam Generator Tube Rupture (Radiological)	100	1650	0%	1650	Yes, Ref. 3.2.5.2, 3.2.5.3
14.2.5	Rupture of a Steam Pipe (Core Response)	0	1650	0%	0	Yes, Ref. 3.2.5.1
	Rupture of a Steam Pipe (Radiological)	0	1650	0%	1650	Yes, Ref. 3.2.5.2, 3.2.5.3
	Rupture of a Steam Pipe (Containment Response)[2]	100	1518.5	2%	1549	Yes, Ref. 3.2.5.10[2]
14.2.6	Rupture of Control Rod Drive Mechanism Housing – RCCA[1] Ejection (Core Response)	0 100	0 1650	0% 2%	0 1683	Yes, Ref. 3.2.5.1 Yes, Ref. 3.2.5.1
	Rupture of Control Rod Drive Mechanism Housing - RCCA[1] Ejection (Radiological)	100	1650	0%	1650	Yes, Ref. 3.2.5.2, 3.2.5.3
14.1.8	Locked Rotor (Radiological)	100	1650	0%	1650	Yes, Ref. 3.2.5.2, 3.2.5.3

(1) Rod Cluster Control Assembly.
(2) Reference is pending NRC review.

FSAR Section 14.2.2, “Accidental Release – Recycle or Waste Liquid,” remains bounded since this release would collect in the Auxiliary Building drains and be held up for release at a later time. This release is administratively controlled by TS. These TS controls are not affected by the MUR uprate.

The steam line break containment integrity analysis was recently reanalyzed by Westinghouse and submitted to the NRC with an associated TS change request on January 11, 2002. This analysis is currently under review with an anticipated approval in July 2002.

3.2.3 Bounding LOCA Related Analyses

All of the LOCA related analyses have been performed at an initial power of 1549 MWt or greater. None of the analyses are changing and the bounding event determinations continue to remain valid for the MUR power uprate.

Table 3.2.3-1 LOCA Related Analyses Bounded by Current Analyses

LOCA Analyses (FSAR Section 14.3)		**Current FSAR Power Assumption**				**Is uprate to 1540 MWt RTP bounded? Reference?**
		% Pwr	**Core MWt**	**Uncer-tainty**	**Power Level**	
14.3.1	Small Break LOCA	100	1650	2%	1683	Yes, Ref. 3.2.5.1
14.3.2	Large Break LOCA	100	1650	2%	1683	Yes, Ref. 3.2.5.1, 3.2.5.3
14.3.3	Core and Internals Integrity Analysis	100	1650	0%	1650	Yes, Ref. 3.2.5.12
14.3.4	Containment Integrity Analysis for LOCA	100	1518.5	2%	1549	Yes, Ref. 3.2.5.3, 3.2.5.19, 3.2.5.16
14.3.5	Radiological Consequences of a LOCA	100	1518.5	2%	1549	Yes, Ref. 3.2.5.3

3.2.4 Other Bounding Licensing Basis Analyses

The NRC RIS listed several other areas that should be addressed by the MUR power uprate submittal. These areas are listed below with reference to either the NRC approval or the appropriate information to show validity of the analysis for the MUR power uprate.

Table 3.2.4-1 Other Bounding Licensing-Basis Analyses

Other Events/Transients	Current Power Assumption				Is uprate to 1540 MWt RTP bounded? Reference?
	% Pwr	Core MWt	Uncer-tainty	Power Level	
Natural Circulation Cooldown	100	1518.5	(1)	(1)	Yes, Ref. 3.2.5.5, 3.2.5.13
Anticipated Transient Without Scram (ATWS)	100	1520	2%	1550	Yes, Ref. 3.2.5.8 [(3)]
Station Blackout	100	1518.5	4.5%	1587	Yes, Ref. 3.2.5.6, 3.2.5.7
Safe shutdown/Appendix R Cooldown	100	1518.5	(1)	(1)	Yes, Ref. 3.2.5.4[(1)]
Flooding	NA[(2)]	NA[(2)]	NA[(2)]	NA[(2)]	Yes, Ref. 3.2.5.17, 3.2.5.18

(1) Evaluation added 8% uncertainty to decay heat. Four percent covered the ANSI/ANS-5.1 decay heat uncertainty while the remaining 4% was used to disposition the 1.4% MUR uprate.
(2) Not dependent on power level.
(3) Study contained generic ATWS analyses with a 2% power increase as a sensitivity study.

3.2.5 References for Section 3.2:

3.2.5.1 NRC SER dated 02/08/2000, Westinghouse 422V+ Fuel Design Change (see reference 3.2.5.9 for license amendment request).

3.2.5.2 NRC SER dated 07/01/1997, Technical Specification Amendments 173 and 177, PBNP Unit 2 Steam Generator Replacement.

3.2.5.3 NRC SER dated 07/09/1997, Technical Specification Amendments 174 and 178, Revised System Requirements to Ensure Post-Accident Containment Cooling Capability.

3.2.5.4 Point Beach Calculation, 97-0118-00-A, "Capability to Achieve Cold Shutdown in Both Units with One CCW pump and Two CCW Heat Exchangers," September 8, 1999.

3.2.5.5 NRC SER dated 11/08/83, "GL 81-21, Natural Circulation Cooldown."

3.2.5.6 NRC SER dated 04/08/92, "Operating License Amendments 130 and 134," regarding CST minimum volume.

3.2.5.7 Point Beach Calculation, Calc N-89-019, Revision 2, "Steam Generator Inventories During One Hour of Station Blackout," 08/30/96.

3.2.5.8 Westinghouse letter to NRC, NS-TSM-2182, "ATWS Submittal," December 30, 1979.

3.2.5.9 NPL 99-0369, "TS Change Request 210, Amendment to facility operating Licenses to Reflect Required Changes to the Technical specifications as a Result of Using Upgraded Fuel," 06/22/99.

3.2.5.10 NMC Letter, NRC 2002-0004, "License Amendment Request 223 Containment Pressure," 01/11/2002.

3.2.5.11 TS 5.5.4, "Radioactive Effluent Controls Program."

3.2.5.12 "Reload Transition Safety Report Point Beach Units 1 and 2 422 Vantage+ Fuel Upgrade/Power Uprate," Westinghouse Report, Rev. 2, November 1999.

3.2.5.13 Calculation WE00005-06, Addendum A (covering mini-uprate), "Natural Circulation Cooldown with One AFW Pump," April 2002.

3.2.5.14 FSAR 14.1.5, Startup of an Inactive Reactor Coolant Loop, June 2001.

3.2.5.15 Letter from Brian Griffin, Westinghouse, to Joseph Presser, WEP, "Overspeed Analysis Report," 10/14/96.

3.2.5.16 WEP 97-522, "Containment Analysis Assuming Reduced Fan Cooler Performance," May 29, 1997.

3.2.5.17 NPC 95-00559 (from NRC to WEPCO), "Review of Individual Plant Examination Submittal for Internal Events – Point Beach Nuclear Plant Units 1 and 2," January 26, 1995.

3.2.5.18 Impact of Instrument Upgrade Power Uprate and Extended Power Uprate Projects on Point Beach Nuclear Plant PRA, R. J. Dremel, Scientech, Revision 0, 10/31/2001.

3.2.5.19 WEP-94-734, "Point Beach Replacement Steam Generator Program Input Assumptions Document – Issue 1," July 26, 1994.

3.3 Analyses Not Bounded for the Proposed MUR Power Uprate

Only two analyses were not bounded by the current licensing basis analyses for the proposed uprated power level: FSAR Chapter 14.2.3, " Accidental Release – Waste Gas," and the radiological environmental parameters. Both are described below.

3.3.1 FSAR 14.2.3, "Accidental Release – Waste Gas"

The PBNP FSAR Chapter 14.2.3 describes the assumptions and conclusions 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 analyses are associated with a core power of 1518.5 MWt. The doses were calculated using the conservative TID-14844 methodology to meet the acceptance criteria of 10 CFR 100 for off-site releases.

Impact of MUR Uprate

The activity for the gas decay tank rupture is made up of long-lived krypton and xenon gases. The original total integrated dose at the nearest site boundary for the gas decay tank rupture is less than 0.8 rem. This value is considerably lower than the 10 CFR 100 limit of 25 rem. A scaling factor of 11 percent for the extended uprate could be applied to this value to conservatively estimate the doses expected for the MUR uprate. Scaling the original dose would result in a site boundary dose of 0.89 rem. This value remains well below the 10 CFR 100 limit of 25 rem.

The volume control tank rupture analysis assumed the release of the long-lived noble gases xenon and krypton and of iodine. The original total integrated dose at the nearest site boundary using the conservative TID-14844 meteorology is less than 0.025 rem for the volume control tank rupture. This is considerably below the limit of 25 rem in 10 CFR 100. The dose equivalent I-131 thyroid dose at the site boundary is less than 0.010 rem. This is also well below the 10 CFR 100 limit of 300 rem. Conservatively applying an 11 percent scaling factor for the extended uprate, the total integrated dose would increase to 0.028 rem and the I-131 thyroid dose would increase to 0.011 rem. These doses remain well below the 10 CFR 100 limits.

The charcoal-filled decay tank rupture assumed the noble gases of krypton and xenon would be released. The site boundary whole body dose resulting from these assumptions is less than 0.1 rem. Bursting of the cryogenic absorber vessel is assumed to release krypton and xenon gases. The calculated site boundary whole body dose is less than 0.03 rem. Both of these are well below the 25 rem limit of 10 CFR 100. Again, an 11 percent scaling factor for the long-lived isotopes can be conservatively applied to disposition the MUR power uprate. This would result in site boundary whole body doses of 0.111 rem for the charcoal-filled decay tank rupture and 0.033 rem for the cryogenic absorber vessel rupture. These scaled doses remain well below the 10 CFR 100 limit of 25 rem.

Conclusions

A conservative scaling factor that equals the percentage of the extended power uprate can be applied to the FSAR documented results of the accidental release of waste gas analyses. Even after applying the scaling factor for the extended uprate, the doses for these accidents remain considerably below the 10 CFR 100 limits of 25 rem for whole body dose and 300 rem for thyroid dose. Based on the considerable conservatisms used in the original analyses, the application of the extended uprate power percentage as a scaling factor, and the considerable margin to the 10 CFR 100 acceptance criteria, the impact on the doses from the MUR power uprate is not significant. It can be concluded that a rupture in the waste gas decay tank, the volume control tank, the charcoal-filled decay tank, or the cryogenic absorber vessel would present no undue hazard to public health and safety. A formal engineering evaluation of the power uprate on the FSAR 14.2.3 accidents will be documented prior to implementation of the MUR uprate and the results incorporated into the next FSAR update using 10 CFR 50.59.

3.3.2 Radiological Environmental Parameters

Original analyses for radiological effluents, dose rates, environmental qualification, and post-accident vital access were all performed assuming a core power of 1518.5 MWt. The radiological impact of an extended power uprate on the following was evaluated:

1) Normal Operation Annual Radwaste Effluent Releases
2) Normal Operation Dose Rates
3) Post-Accident Dose Rates
4) Post-Accident Access to Vital Areas

An engineering evaluation for the impact of an extended power uprate of up to 10.5 percent to 1678 MWt core power was performed. For normal operation dose rates and shielding, a core power of 1678 MWt was used while the impact from exposure to the primary coolant was based on a core power level of 1683 MWt. This additional conservatism was referred to as an 11 percent uprate in the evaluations. The radwaste effluent assessment assumed a core power level of 1683 MWt, but used flow rates and coolant masses at the full NSSS power level of 1684 MWt (relating to a core power of 1678 MWt). These core powers will bound the MUR power uprate to 1540 MWt since the increase in the radiological dose is typically proportional to the percent increase in power.

No equipment changes are necessary to support these analyses. However, certain environmental qualification (EQ) documentation needs to be revised for the new uprated environmental parameters. Post-LOCA vital access must also be addressed for the uprated power.

3.3.2.1 Normal Operation Annual Radwaste Effluent Releases

Liquid and gaseous effluents released to the environment during normal plant operations contain small quantities of radioactive materials. Therefore, the design of the PBNP radioactive waste disposal system is based on meeting the requirements of 10 CFR 20 dosage level guidelines. The radwaste system is operated to ensure the design objectives of 10 CFR 50, Appendix I are met. On June 4, 1976, Wisconsin Electric Power Company (WEPCO, then the licensee) submitted to the NRC Amendment 20 to the PBNP Final Facility Description and Safety Analysis Report (FFSDAR). Amendment 20 became Appendix I to the PBNP FSAR. The analyses presented in the PBNP FSAR Appendix I demonstrate that the PBNP radwaste system can be operated such that impact of effluent releases from the site are within the design objectives of 10 CFR 50, Appendix I. Technical Specifications are also in place which require that PBNP maintain liquid releases below ten times 10 CFR 20 concentration values and offsite doses conforming to 10 CFR 50, Appendix I. Similarly, gaseous effluent Technical Specifications limit the dose rate as well as require offsite doses to conform to 10 CFR 50, Appendix I.

Power uprate will increase the activity level of radioactive isotopes in the reactor coolant and the secondary-side. Due to leakage or process operations, fractions of these fluids are transported to the liquid and gaseous radwaste systems where they are processed prior to discharge. As the activity level in the reactor coolant and secondary side are increased, the activity levels of the radwaste inputs are proportionally increased. The impact of the power uprate on the ability of radwaste system to maintain releases within the PBNP licensing basis is evaluated.

Radiation Source Terms

In evaluating the impact of the power uprate on radwaste effluents, the methodology in Draft Regulatory Guide 1.BB and NUREG 0017 (References 3.3.2.5.1 and 3.3.2.5.2) was used to establish the relative change in expected reactor coolant activities as well as a comparison of the original input parameters to the power uprate parameters. The evaluation demonstrated that the limiting increase in the reactor coolant source for long-lived isotopes is the percentage of the analyzed power uprate (rounded to 11 percent for this evaluation). For isotopes with short half-lives, the range of the increase is from the percentage of uprate (11 percent) to approximately 21 percent.

Liquid and Gaseous Releases

The increase in the activity concentration of isotopes with long half-lives due to power uprate is directly proportional to the change in power level. There was an approximate 11 percent increase in the liquid effluent release concentrations, as this activity is based on RCS activity possessing long half-lives. Tritium releases in liquid effluents increase proportionately to the evaluated power uprate. Since the maximum increase due to the power uprate, relative to the liquid releases, is approximately 11 percent, the increase in the estimated dose consequence evaluated in PBNP FSAR Appendix I will also be bounded by this value. Sufficient margin exists between the current stated value and the design objective for liquids. Therefore, the liquid radwaste effluent treatment system design remains capable of maintaining normal operation offsite releases within the requirements of the PBNP current licensing basis.

For the gaseous releases, the evaluation concluded that gaseous releases for Kr-85 are limited to an 11 percent increase directly proportional to the increase in production dictated by the 11 percent power increase and its long half-life. For other noble gases, the release is scaled on coolant activity. Isotopes with shorter half-lives have increased releases, compared to the power uprate. Basing the incremental iodine, noble gas and tritium release on the change in reactor coolant concentrations caused the releases to increase approximately 11 percent for long-lived isotopes and up to 21 percent for short-lived isotopes. The other components of the gaseous release (i.e., particulates in the building ventilation systems and water activation gases) are not impacted by power uprate using the methodology of draft Regulatory Guide 1.BB and NUREG 0017.

With respect to the gaseous pathway, for beta and gamma air doses, assuming that the change in doses will be proportional to the change in beta and gamma energy flux, yielded a 13 percent increase in beta doses and a 16 percent increase in gamma doses. Estimated increased radioiodine releases of 12 percent result in a proportionate increase in thyroid dose. Sufficient margin exist between the current stated value and the design objective for gaseous releases. Therefore, the gaseous radwaste effluent treatment system design remains capable of maintaining normal operation offsite releases within the requirements of the PBNP current licensing basis.

Solid Waste

Regulatory guidance for a "new" facility estimates the volume and activity of solid waste as being linearly related to the core power level. However, for an existing facility that is undergoing power uprate, the volume of solid waste would not be expected to increase proportionally. This is because the power uprate neither appreciably impacts installed equipment performance nor requires drastic changes in system operation. Only minor, if any, changes in waste generation volume are expected. It is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity (i.e., 11 percent).

In conclusion, the overall volume increase of waste generation resulting from uprate is expected to be minor. However, the long-lived activity contained in the waste is expected to be proportional to the percentage of uprate. There are no acceptance criteria for solid waste except that disposal of solid radioactive waste must be done with regard to several federal regulations. The PBNP Process Control Program provides the guidance for analyzing, processing, and packing of radioactive wastes in order to produce a final waste form that is acceptable for transportation and burial at a licensed radioactive waste disposal site. This program does not change as a result of the uprate. Solid waste processing and disposal will continue to be controlled by the above program to maintain compliance with regulatory requirements for the power uprate to 11 percent.

Conclusion

The impact of an extended power uprate on the estimated annual radwaste effluent releases/doses is limited to less than 16 percent as determined in the engineering evaluation. Based on the estimated fractions of the 10 CFR 50, Appendix I design objectives for the extended uprate condition, the liquid and gaseous radwaste system effluent treatment design remains capable of maintaining normal operation offsite releases and doses within the requirements of the PBNP licensing basis. The estimated effluents for the MUR uprate would be substantially less. The activity of solid waste could increase proportionately with the power uprate but the increase in waste generation volume is expected to be minor for an extended uprate. Solid waste will continue to be controlled by the Process Control Program at PBNP to maintain compliance with regulatory requirements.

3.3.2.2 Normal Operation Dose Rates

The impact of the extended power uprate on the normal operation dose rates and the adequacy of existing shielding was evaluated to ensure continued safe operation within the regulatory limits. In addition, the impact of the extended power uprate on the normal operation component of the total integrated dose used for environmental qualification of equipment for radiation was evaluated.

The evaluation for normal operation dose rates compared the original design basis source term to the extended power uprate source term (1683 MWt core). The approach estimated the scaling factor impact rather than developing actual dose rate and integrated dose estimates at the various environmental zones in the plant. The use of scaling factors is a generally accepted practice in the industry and application of scaling factors is not considered a change in methodology. The scaling factor was used to determine the personnel exposure levels, the adequacy of shielding, and the acceptability of safety-related equipment for radiation within the specified environmental zones. The acceptance criteria for personnel exposure and the adequacy of the existing shielding are the requirements of 10 CFR 20. However, for EQ, the results of this analysis (i.e., the radiation levels) do not have acceptance criteria, but rather become the normal radiation specification requirements for the PBNP EQ Program. This section discusses the evaluation and the impact on personnel exposure and environmental qualification.

A core uprate will increase the activity inventory of fission products in the core by approximately the percentage of the power uprate for normal operation. Therefore, an increase in radiation levels during normal operation would be 10.5 percent for a 10.5 percent uprate to 1683 MWt. For the smaller MUR uprate, the increase in radiation levels during normal operation is expected to be much smaller and comparable to the percent increase in power (approximately 1.4 percent). The radioactivity levels in the primary coolant, secondary coolant, and other radioactive process systems and components will also be impacted. The core radionuclide inventory evaluated is based on 1683 MWt with one percent failed fuel. The evaluation was for a full 10.5 percent uprate and, therefore, bounds the proposed 1.4 percent MUR uprate.

Uprate Impact on Normal Operation Dose Rates: Personnel Exposure and Shielding

The increase in expected radiation levels due to an extended uprate will not significantly affect radiation zoning or shielding requirements in various areas of the plant. It is expected that the increase due to the uprate will be offset by the conservatisms in the original design basis source terms used to establish the radiation zones, the plant Technical Specifications that limit RCS concentration levels to well below the design levels, and the conservative analytical techniques typically used to establish shielding requirements. The increase in radiation levels due to a 1.4 percent MUR uprate would be expected to be much smaller.

Regardless of the above, individual worker exposures will be maintained within acceptable limits by the site ALARA program that controls access to radiation areas and by procedural controls that may be used to compensate for increased radiation levels.

Uprate Impact on Normal Operation Dose Rates: Equipment Qualification

For the contribution to the EQ normal dose for components located inside containment and within the primary shield walls (i.e., near the reactor vessel), the existing design basis environmental parameters bound the impact of uprate. No changes are required to equipment or equipment-specific environmental qualification documentation.

The area located outside the primary shield wall but inside the secondary shield walls could potentially be affected by increased normal operating doses. This potential increase is expected to be approximately 15 percent. Equipment in this area was reviewed for acceptability by the PBNP. A review of the EQ program and the appropriate equipment qualification summary sheets (EQSSs) found that all components continue to have sufficient margin at the extended uprate conditions.

Environmental qualification normal operating doses for equipment located outside the secondary shield walls also will not change. It was determined that the normal operation dose is insignificant compared to the accident dose. No changes are required to equipment or equipment-specific environmental qualification documentation.

The current environmental conditions noted for areas outside of containment remain bounding for the extended uprate. No changes are required to equipment or equipment-specific environmental qualification documentation.

Conclusions for Normal Operating Doses

The impact of the extended power uprate on the normal operation dose rates and the adequacy of existing shielding was evaluated to ensure continued safe operation within the regulatory limits. It was determined that the increase in expected radiation levels due to an extended uprate will not significantly affect radiation zoning or shielding requirements in various areas of the plant. Additionally, individual worker exposures will be maintained within acceptable limits by the site ALARA program. Evaluation of the uprated normal operating doses for EQ of equipment found that all equipment maintained sufficient margin. No changes are required to equipment as a result of the extended uprate. Environmental Qualification documentation will require revision for equipment outside the primary shield but inside the secondary shield walls inside containment and for the general normal radiation basis documents. This evaluation bounds the MUR uprate.

3.3.2.3 Post-Accident Dose Rates

Power uprate impacts the equilibrium core inventory and, therefore, the post-accident radiological source terms. The impact was evaluated based on a comparison of the original design basis source terms to the power uprate source terms developed by Westinghouse for a 1683 MWt core power. The approach was to estimate impact by developing scaling factors based on the source term comparison rather than developing actual integrated dose estimates at the various zones or component specific locations. There are no acceptance criteria for these evaluations. Rather, the results become the revised accident radiation specification requirements for the PBNP Environmental Qualification (EQ) Program for an extended power uprate of 10.5 percent.

Post-LOCA Radiation Environment Inside Containment

The accident gamma radiation dose inside containment as a function of time was developed in WCAP 7410-L, "Topical Report Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)," December 1970, and is currently documented in PBNP FSAR Figure 14.3.4-15. The WCAP supported the early efforts addressing equipment qualification on a generic basis for Westinghouse plants. The most limiting condition for any Westinghouse plant design was used. This included conservative assumptions relative to containment volume, the estimated post-accident radiological release into containment, and took no credit for in-containment shielding.

The Westinghouse model for development of FSAR Figure 14.3.4-15 was used to develop the containment accident dose for the uprated condition. The graph developed for the uprated conditions was then super-imposed on the FSAR figure to demonstrate that the existing curve bounds the uprate curve. Figure 3.3-1 of this attachment contains the comparison and shows that even for a 10.5 percent extended uprate in core power, the original curve from the Westinghouse WCAP remains bounding. The MUR uprate is bounded by the extended uprate.

The core uprate does not affect the estimated beta dose in containment for the EQ Program at PBNP. The EQ Program is based on the IE Bulletin 79-01B/DOR Guidelines (Reference 3.3.2.5.3) that conservatively assumed the beta dose inside of containment to be 2.0 E 08 Rads. The DOR guideline value is independent of power level.

Post LOCA Radiation Environment Outside Containment

The uprated core inventory is used to develop the post-LOCA gamma integrated energy releases per energy group versus time for pressurized recirculating fluid. Both shielded and unshielded cases were reviewed. For the unshielded case, the scaling factor impact on post-accident gamma dose rates due to uprate was estimated by ratioing the gamma energy release rates as a function of time for the uprate power level. Although the unshielded case does not address shadow shielding or concrete floors and walls, the unshielded case does consider the effect of the steel wall (0.375 inches) of the pipe surrounding the fluid. For the shielded case, the original and uprated source terms were weighted by the concrete reduction factors for each energy group. The maximum unshielded/shielded power uprate gamma dose scaling factor for the post-LOCA radiation environment was estimated to be 1.18.

For beta radiation outside of containment, PBNP maintains its position that the beta radiation is effectively attenuated to negligible values because the radioactive fluid is completely contained within stainless steel piping. This position is not impacted by uprate.

Impact on Equipment Inside and Outside Containment

The pre-uprate accident environmental conditions inside of containment for both gamma and beta radiation are bounding for a 10.5 percent uprate. No changes are necessary to equipment or to environmental qualification documentation.

Outside the containment the current beta radiation conditions post-accident remain bounding. However, the post-accident gamma radiation outside of containment will increase approximately by a factor of 1.18 for a 10.5 percent uprate. The effect of this increase on equipment was evaluated for equipment qualification. Components identified outside of containment that are affected by the factor of 1.18 increase in gamma dose were found to contain adequate margin with the exception of Okonite butyl rubber insulated cable for the safety injection pump motors. Preliminary investigations indicated that type testing has been completed at conditions bounding the extended uprate power conditions at PBNP. Point Beach Nuclear Plant is confident it can resolve this issue prior to implementation of the MUR uprate license amendment. The Design Guideline for Environmental Qualification Service Conditions (DG-G11) and the appropriate EQSSs will be revised to reflect the uprate condition.

Conclusions

Post-accident radiation doses were evaluated for a 10.5 percent uprate. Equipment continues to meet the uprated post-accident conditions both inside and outside of containment with the exception of the butyl rubber insulated cable. Point Beach Nuclear Plant is confident it can qualify the use of the butyl rubber cable at the new post-accident radiation levels and commits to having this qualification and other EQ documentation revisions for extended uprate conditions completed by implementation of the MUR power uprate.

3.3.2.4 Vital Access

In accordance with NUREG 0578 (Reference 3.3.2.5.4), Item 2.1.6.b and NUREG 0737 (Reference 3.3.2.5.5), II.B.2, vital areas are areas within the plant that require access or occupancy to support accident mitigation or recovery following a loss of coolant accident (LOCA). Both NUREGs limit the allowable operator dose while performing vital access functions to 5 Rem. PBNP submitted its implementation plan for NUREG 0578, Item 2.1.6.b to the NRC on December 31, 1979. During 1980 and 1981, PBNP submitted revised mission dose evaluations to the NRC. The revised doses used core inventories developed using the NUREG 0578 source terms. Following 1981, subsequent assessments were made relative to general comparisons between NUREG 0578 and NUREG 0737 source terms. The NRC accepted the PBNP NUREG Item II.B.2 response on November 3, 1983.

The impact of a 10.5 percent power uprate on the accident radiation doses received by operators in vital areas during post-LOCA conditions is evaluated based on a comparison of the original design basis source terms to the power uprate source terms. Scaling factors were developed using the new core inventory rather than actual dose rate estimates. The scaling factors were developed based on both NUREG 0737 and NUREG 0578 since the existing licensing bases referenced both NUREGs. The more limiting scaling factor was applied as appropriate. The maximum accident dose rate scaling factor based on NUREG 0578 source terms in the recirculating fluid was projected to be 1.21. The maximum accident dose rate scaling factor based on NUREG 0737 was 1.17. Additionally, changes to containment sump water volume had occurred since the original analysis. This change was evaluated under current operating conditions when the sump volume change occurred and was found to meet the acceptance criteria. Because scaling factors used in the extended uprate evaluation were developed from the original analyses, a sump water level volume adjustment factor of 1.303 was also applied to the scaling factors.

Uprate Impact on Vital Access

The accumulated dose for vital access for a Unit 1 and Unit 2 accident increases by approximately a factor of 1.6 based on the combined effect of the extended uprate dose rate and sump volume scaling factors. This factor compares uprated conditions to the original licensing basis. The review concluded that no target area that was previously accessible has become inaccessible due to uprate with the exception of the Unit 1 sample room (for a Unit 1 LOCA) and the Unit 2 sample room (for the Unit 2 LOCA). This issue has been captured by the PBNP corrective action process and will be resolved prior to implementation of the MUR uprate.

The accumulated dose while collecting a post-accident coolant sample for a Unit 1 and Unit 2 accident was also rescaled using the 1.6 factor for uprated conditions. Operator exposure while performing coolant sampling following a Unit 1 event will slightly exceed regulatory limits following extended uprated conditions. However, this assessment is for a full uprate of 11 percent. Scaling factors were developed for the MUR power uprate to address this issue. It was found that for the MUR power uprate, the Unit 1 coolant sampling following a Unit 1 event will remain within the regulatory limit of five (5) Rem. Although no further work was necessary for the MUR power uprate, the extended power uprate issue for the sampling room is being addressed in the same corrective action request as referenced.

The accumulated dose while performing operations at the Unit 1B32 Motor Control Center (MCC), the Unit 2B32 MCC, and the C59 Waste Disposal Control Panel for a Unit 1 or Unit 2 accident increased by approximately a factor of 1.52 for uprated conditions when compared to the original licensing basis. This scaling factor is based on the combined NUREG 0737 scaling factor and the sump volume adjustment factor. The review concluded that all of the target areas continued to remain accessible for uprated conditions. However, it was noted that the referenced accumulated doses for these areas do not address the dose contribution received during transit. Although doses are not exceeded for these areas for the extended uprate based on the original licensing basis (target area dose only with no transit contribution), PBNP is performing work to address the transit time doses through the plant's corrective action program. This issue will also be resolved prior to implementation of the MUR power uprate.

Conclusions

The evaluation performed for the extended power uprate identified issues with vital access. These issues included post-accident sampling doses and transit doses to three target areas. The actions stated above to resolve the Vital Access issues will be completed prior to the implementation of the MUR uprate license amendment.

3.3.2.5 References

3.3.2.5.1 Draft Regulatory Guide 1.BB, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," 02/28/75.

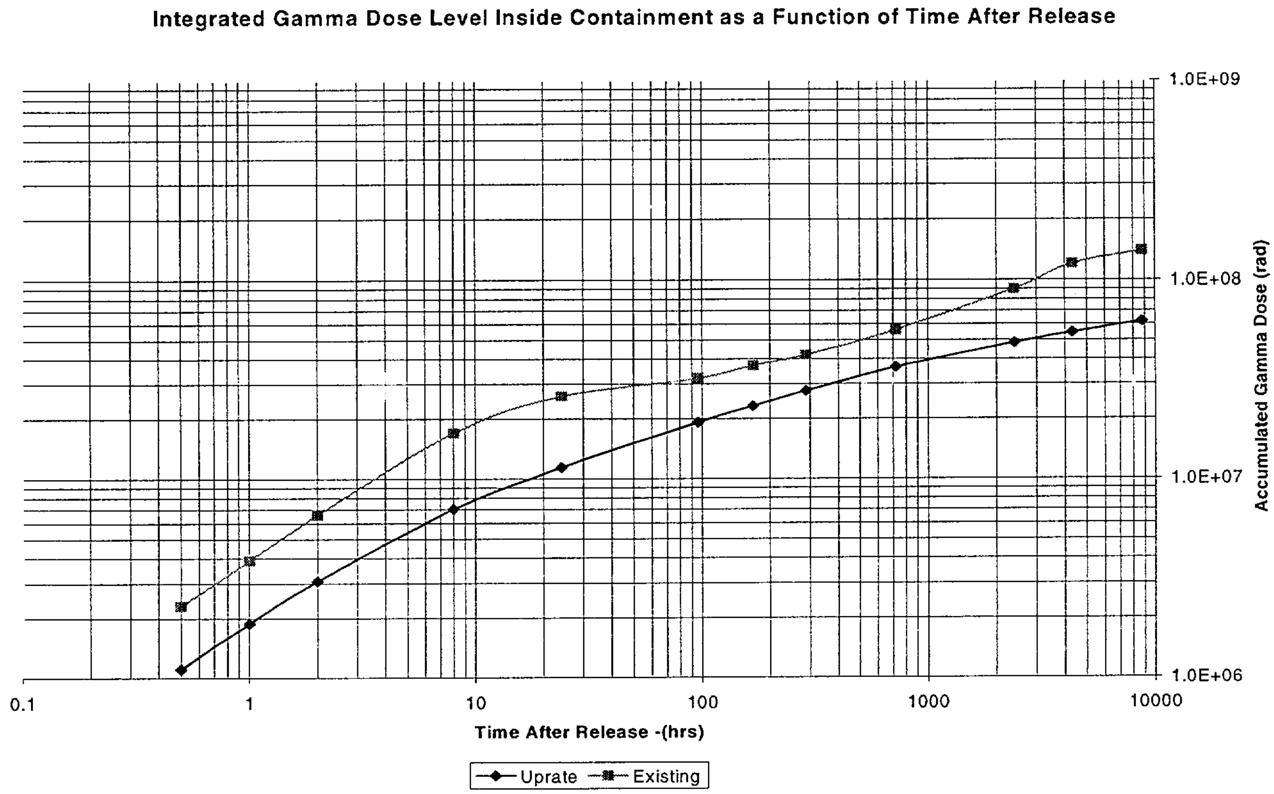
3.3.2.5.2 NUREG 0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," April 1976.

3.3.2.5.3 IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," including Enclosure 4 (DOR Guidelines), "Guidelines for Evaluating Environmental Qualification of Class 1E Electric Equipment in Operating Reactors," January 14, 1980.

3.3.2.5.4 NUREG-0578, "TMI Lessons Learned Task Force Status Report and Short Term Recommendations," July 1979.

3.3.2.5.5 NUREG 0737, "Clarification of TMI Action Plan Requirements," November 1980.

FIGURE 3.3-1
Impact of Power Uprate on Post-LOCA Radiation Environment Inside Containment



3.4 Mechanical/Structural/Material Component Integrity and Design

The following sections discuss the changes to NSSS design parameters for the MUR uprate and the evaluation of the continued structural integrity of major plant components, reactor vessel integrity, and component inspection and testing programs. This section did take into account the NRC content guidance of RIS 2002-03.

3.4.1 NSSS Design parameters

The NSSS design parameters developed by Westinghouse are the fundamental NSSS parameters used as primary input to the NSSS system and component analyses, design transient analysis, and FSAR Chapter 14 safety analyses. The parameters are established using conservative assumptions in order to provide bounding conditions appropriate for analyses. No uncertainties are incorporated into the values of the parameters.

NSSS component and system analyses were performed for both 1518.5 MWt and for an uprated power of 1650 MWt during the Unit 2 Replacement Steam Generator (RSG) project. Note that this analysis is applicable to both PBNP units. In 1999, the NSSS design parameters were redeveloped for the change to the Westinghouse 422V+ fuel (fuel upgrade) and again included the uprated power of 1650 MWt (Reference 3.4.7.18, 3.4.7.19). The NSSS design parameters for the fuel upgrade evaluated the accident analysis at 1650 MWt, as seen in the previous section. These projects used uprated design parameters to encompass future power uprates.

Although no design transient, component, system, or accident analyses were performed specifically for the MUR uprate (1540 MWt), design parameters were developed at 1540 MWt for comparison. Table 3.4.1-1 compares the design parameters for the current power level to the parameters for a power uprate to 1540 MWt. The 1540 MWt MUR power uprate is bounded by the analyses performed at 1650 MWt during both the RSG and fuel upgrade projects.

Table 3.4.1-1, Design parameters for uprated powers

Parameter	**1518.5 MWt**	**1540 MWt**	**1650 MWt**
Core power	1518.5 MWt	1540 MWt	1650 MWt
NSSS power	1524.5 MWt	1546 MWt	1656 MWt
RCS Pressure	2000 or 2250 psia	2250 psia	2000 or 2250 psia
Tavg range	557.0°F – 574.0°F	558.1°F – 574.0°F	559.4°F – 578.7°F
Thermal design flow	89,000 gpm	89,000 gpm	85,200 gpm
SG tube plugging	0 to 10 percent	0 to 10 percent	0 to 25 percent
Steam Pressure	612 psia - 857 psia	664 psia – 800 psia	612 psia – 803 psia
Steam Temperature	488.7°F – 526.2°F	497.3°F – 518.2°F	485.5°F – 518.7°F
Steam Flow Rate (lbm/hr)	6.55 E06 lbm/hr to 6.60 E06 lbm/hr	6.72 E06 lbm/hr to 6.75 E06 lbm/hr	7.22 E06 lbm/hr to 7.26 E06 lbm/hr
Reference	Ref. 3.4.7.15 and 3.4.7.18	Ref. 3.4.7.16	Ref. 3.4.7.15

3.4.2 Mechanical/Structural/Material Component Integrity

The existing analyses of record were performed during the Unit 2 Replacement Steam Generator (RSG) Program and remains valid for the MUR power uprate. The RSG component analyses were performed using the NSSS design parameters at 1650 MWt. During this evaluation, the component designers reviewed the original design information related to the components. This review included stress and fatigue effects. The evaluations performed confirmed the components continued to satisfy the applicable codes, standards, and regulatory guides at the uprated conditions. Stresses and cumulative usage factors (CUF) remained below allowable limits. Table 3.4.2-1 below summarizes the bounding component analyses. It includes reference to the component, power level assumed, and the acceptance criteria to establish validity of the analysis of record (AOR). Additionally, the table references the PBNP project for which the analysis was specifically performed.

Table 3.4.2-1, Bounding AOR for NSSS Components

Component	Core Power (MWt)	Acceptance Criteria for AOR	Project
Reactor Vessel, Nozzles, Supports	1650	Unit 1: Ref. 3.4.7.1, Ref. 3.4.7.3 for faulted conditions Unit 2: Ref. 3.4.7.2, Ref. 3.4.7.3 for faulted conditions	RSG
Reactor Core support structures and vessel internals	1650	Evaluated to Westinghouse internal criteria similar to Ref. 3.4.7.13. Internals were designed prior to introduction of Reference.	RSG
Control Rod Drive Mechanisms	1650	Ref. 3.4.7.4	RSG
NSSS piping: Reactor coolant loop piping Primary Equipment Nozzles	 1650 1650	 Ref. 3.4.7.5 Ref. 3.4.7.5	RSG
NSSS piping supports	1650	Ref. 3.4.7.9 as supplemented with Ref. 3.4.7.10 and 3.4.7.11	RSG
Other NSSS Fluid system piping: Chemical and Volume Control Residual Heat Removal Safety Injection and Containment Spray System Sampling System Component Cooling System	1650	All Ref. 3.4.7.5	RSG
BOP piping (NSSS interface systems) Main Steam Condensate and Feedwater Auxiliary Feedwater Steam Generator Blowdown	1650	All Ref. 3.4.7.5	FSAR

Table 3.4.2-1, Continued

Unit 1, Model 44F Steam generators: Tubes Internal support structures Shells Nozzles	1650	Ref. 3.4.7.8 for Shell and original equipment. Ref. 3.4.7.8 and 3.4.7.14 for replacement SGs.	RSG
Unit 2, Model Δ47 Steam generators: Tubes Internal support structures Shells Nozzles	1650	Ref. 3.4.7.8 for original SG. Ref. 3.4.7.6 and 3.4.7.8 for RSGs.	RSG
Reactor Coolant Pumps: Structural Electrical	1650	Ref. 3.4.7.7, Ref. 3.4.7.12 used only to demonstrate accept-ability even though the PBNP RCPs pre-date the inclusion of pumps into ASME code.	RSG
Pressurizer: Shell Nozzles Surge line	 1650 1650 1650	 Ref. 3.4.7.8 Ref. 3.4.7.8 Ref. 3.4.7.6, Ref 3.4.7.5	RSG
Safety-Related Valves	1650	Ref. 3.4.7.15	RSG

3.4.2 Flow Induced Vibration (FIV)

The impact of an 8.7 percent uprate (to 1650 MWt) on reactor vessel internals susceptibility to FIV was evaluated during the RSG project. This analysis is documented in WCAP 14459, "Reactor Pressure Vessel And Internals System Evaluations For The Point Beach Units 1 And 2 Power Uprating/Replacement Steam Generator," 1996 (Reference 3.4.7.17). This report found the structural integrity of the PBNP reactor internals to remain acceptable with regard to flow-induced vibrations.

3.4.3 High Energy Line Break Locations and Jet Impingement/Thrust Forces

High energy lines are defined by PBNP FSAR Appendix A.2 as lines where the combined pressure and temperature of the fluid exceeds 275 psig and 200°F during normal operation. The following piping outside of containment meets these conditions and requires analysis:

a) Main Steam piping,
b) Turbine Bypass to Condenser,
c) Auxiliary Steam Supply to Auxiliary Feedwater Pump Turbine,
d) Auxiliary Steam Supply to Waste Disposal Equipment,
e) Steam Generator Blowdown Piping,
f) Main Feedwater Piping,
g) Sample Lines.

The PBNP FSAR A 2 states that visual inspections of the piping and surrounding areas were performed for the above piping systems. Based on these inspections it was found that no safeguard equipment was located in the areas traversed by the piping of items e, f, and g. The inspections also determined that wall barriers and drainage were adequate to handle environment or flooding effects resulting from a pipe failure in these areas. Therefore, these three systems did not required detailed analysis. There will be no modifications to any of these systems or to equipment in these areas as a result of the MUR power uprate. Therefore, the current FSAR Appendix A.2 documentation remains unchanged and is bounding for the MUR uprate.

The remaining lines (a through d) were also assessed for the MUR power uprate. There will be no modifications to these systems or any new lines added to these systems. Therefore, no new high-energy lines will be created for these systems with the proposed MUR uprate. Mass and energy releases could potentially increase due to the increased core power and decay heat. However, a review of the mass and energy releases used for the environmental effects of HELBs outside of containment showed adequate margin to bound the MUR power uprate. The mass and energy releases used at PBNP are generic releases based on a four-loop Westinghouse plant. Conservative assumptions were made in determining these mass and energy releases to ensure early tube bundle uncovery and the earliest superheat initiation time. The generic releases have been compared with plant specific releases from a 1650 MWt two-loop Westinghouse plant and a 3250 MWt four-loop Westinghouse plant. The generic releases will remain bounding for the MUR power uprate to 1540 MWt, since they are conservative when compared to plants with larger cores.

The effects of pipe whip and jet impingement are generally considered to be a direct function of pressure. The changes in the operating pressure of the systems listed above due to the MUR power uprate are insignificant. The system pressure of items a, b, and c above either remains at the current operating level or decreases slightly because the full power operating main steam pressure decreases with the MUR power uprate. These three systems were evaluated with the conservative assumption that the plant was at maximum hot standby conditions (1020 psig and 545°F). This condition yields the highest operational steam system pressure. Hot standby condition will not change when the MUR uprate is implemented. Analysis for item d, the Auxiliary Steam Supply to the Waste Disposal Equipment, is based on maximum operating conditions for that system as listed in FSAR Appendix A.2. These conditions are an operating pressure of 900 psig and an operating temperature of 534°F. The operating pressure and temperature will also decrease slightly due to the decrease in the main steam full power operating pressure with the MUR uprate. Therefore, there is no impact to pipe whip and jet impingement concerns for any of these lines.

3.4.4 Reactor Vessel Integrity

The Code of Federal Regulations specifies requirements for operation of reactor vessels. These requirements include criteria for heatup and cooldown limit curves, low temperature overpressure protection, pressurized thermal shock, upper shelf energy, and reactor vessel radiation surveillance capsule monitoring and testing. Operation of the reactor vessels is contingent upon satisfying criteria for each for these requirements. These reactor vessel integrity criteria at PBNP were evaluated for the impact of the MUR power uprate. The evaluation is described below.

Neutron Irradiation

The revised design conditions of the power uprate can affect neutron irradiation analyses generally in two ways. One way is that changes in T_{cold} may affect the value used in the fluence calculations. The second way is that the increase in core power will increase the neutron fluence experienced by the vessel.

The current neutron fluence evaluations assume that T_{cold} is maintained between 532°F and 552°F. The post-MUR power uprate T_{cold} is calculated to be between 528°F and 544.5°F. Actual T_{cold} temperatures lower than that assumed in the analysis (e.g., 528°F) is conservative as a colder moderator will attenuate the neutron flux and reduce the fluence on the reactor vessels. The 544.5°F value is within the assumed T_{cold} temperature band. Thus, the T_{cold} temperature changes as a result of this power uprate do not adversely impact current fluence evaluations.

Increased neutron fluence experienced by the reactor vessel is discussed in the following sections of this evaluation.

Heatup and Cooldown Curves

Allowable neutron fluence values for the reactor vessels are predicted and controlled in the development and use of the Pressure-Temperature Curves (P-T Curves). These P-T Curves, or heatup and cooldown curves, utilize fluence predictions and fracture mechanics data from vessel materials to define allowable steady state and heatup and cooldown pressure–temperature operating limits. PBNP utilizes a single P-T Curve for both units that is located in the Pressure Temperature Limits Report (PTLR), TRM 2.2.

The heatup and cooldown curves were determined using the methods of Appendix G to the ASME Boiler and Pressure Vessel Code III and the irradiated material properties of the reactor core region materials. The effects of irradiation on the core region materials were estimated by using post-irradiation test data of the specimens contained in the integrated reactor vessel irradiation surveillance capsule program.

The calculation determined the estimated fluence values at the end of the operating license. These fluence values are then used to calculate the projected Adjusted Reference Temperature (ART) values. The ART values define the shape and limits of the curves. The curves are valid until the neutron fluence values are exceeded. The current P-T curves are licensed to the following limits:

Unit 1 – Effective through 25.59 EFPY
Unit 2 – Effective through 30.51 EFPY

The Unit 1 value corresponds to a fluence value of 2.25×10^{19} n/cm^2 at the inside surface of the limiting component (Intermediate to Lower Shell Circ Weld).

The Unit 2 value corresponds to a fluence value of 2.49×10^{19} n/cm^2 at the inside surface of the limiting component (Intermediate to Lower Shell Circ Weld).

The proposed power uprate will slightly change the rate of neutron flux to the reactor vessel. The P-T curves 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 the MUR 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.

Low Temperature Overpressurization (LTOP)

The LTOP setpoints for PBNP are also located in the Pressure Temperature Limits Report (PTLR) and use identical fluence values as the heatup and cooldown curves. As such, the LTOP values have the same applicability limits of EFPY and fluence as do the heatup and cooldown curves.

Similar to the P-T curves, the LTOP setpoints will be revised using the existing processes prior to exceeding the above stated fluence and EFPY values. Therefore, the proposed 1.4 percent power uprate has no adverse impact upon existing LTOP setpoints.

Pressurized Thermal Shock (PTS)

PTS is an event that results in a rapid and severe cooldown in the primary coolant system coincident with a high or increasing primary system pressure. The NRC has issued a formal rule (10 CFR 50.61) on PTS that established screening criteria on PWR vessel embrittlement, as measured by the maximum nil ductility reference temperature in the limiting beltline component, termed RT_{PTS}. These RT_{PTS} screening values were set by the NRC for beltline axial welds, forging or plates, and for beltline circumferential weld seams for plant operation to the end of plant license.

The limiting RT_{PTS} value for both units is the intermediate to lower shell circumferential weld. The 10CFR50.61 limit for circumferential welds is 300°F. The Unit 1 weld (SA-1101) is calculated at 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 MUR uprate.

PBNP RT_{PTS} values will remain within regulatory limits following the 1.4 percent power uprate.

Upper Shelf Energy (USE)

The lower upper shelf Charpy energy concern is associated with the determination of acceptable reactor pressure vessel (RPV) toughness during plant operation when the vessel is exposed to additional irradiation.

The requirements on upper shelf energy are included in 10CFR50, Appendix G. This document requires utilities to submit an analysis at least three years prior to the time that the upper shelf energy of any of RPV material is predicted to drop below 50ft-lb, as measured by Charpy V-notch specimen testing. Since the PBNP upper shelf energy (USE) value at 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 report specifically evaluated the reactor vessel fluence at the end of the license renewal period taking into account extended power uprate conditions. The estimated fluence at End of Life Extension (EOLE) was 51.9 EFPY (Unit 1) and 53.6 EFPY (Unit 2). 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. This analysis will be updated prior to PBNP nearing the EFPY values used in the USE analysis. Engineering monitors EFPY values monthly through review of the PBNP Monthly Operating Report. This will address any increase in fluence due to the MUR power uprate.

As outlined above, the 1.4 percent power uprate has no adverse impact upon USE evaluations.

Surveillance Capsule Monitoring and Withdrawal Schedule

The original PBNP surveillance program was prepared in accordance with ASTM E 185-66 and consisted of six surveillance capsules in each unit that were attached to the outside of the reactor internals thermal shield. Each capsule contained mechanical specimens, dosimetry, and thermal monitors. The mechanical specimens were fabricated from material representative of the reactor pressure vessels. All surveillance capsules (except for standby capsules) have been removed and tested.

The actual heats of the limiting weld metals were not included in the PBNP capsules. Therefore, PBNP is a member in the B&W Owners Group Materials Sub-committee 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.

In addition, a new PBNP surveillance capsule will be installed during the Spring Unit 2 refueling outage for the purpose of obtaining relevant fracture toughness data at the End of Life Extension (EOLE) fluence. The new PBNP surveillance capsule contains surveillance specimens that will be used to directly measure the fracture toughness of the limiting PBNP weld metal heats.

The target fluence for the supplemental surveillance materials will correspond to the peak reactor vessel fluence at EOLE and considers the affects of a 1.4 percent power uprate. The projected EOLE fluence value for the capsule (installed in Unit 2) is 5.085×10^{19} n/cm^2. Based upon current fluence data, it is projected that this capsule be withdrawn in the year 2022. Surveillance data obtained from this capsule will provide direct fracture toughness measurements for the limiting weld metal near the maximum fluence at EOLE. This data will provide direct evidence to validate previous reactor vessel life assessments and a measure of the actual margins available for the PBNP RPVs.

The PBNP peak reactor vessel fluences and the fluence for all supplemental surveillance capsules will be re-evaluated in the future to reflect actual reactor operation. As further reactor operation occurs, better vessel and capsule fluence estimates can be made and a more definitive capsule withdrawal schedule will be established. Because fluence calculations will be periodically revised to include further test data and to determine the surveillance capsule withdrawal schedules, the 1.4 percent power uprate has no adverse impact upon the PBNP Surveillance Capsule Program.

3.4.5 Component Inspection and Testing Programs

The 1.4 percent MUR power uprate will not require any plant modifications to components in the in-service inspection (ISI) program or the in-service testing (IST) program. Additionally, none of the changes in the design parameters warrant changes to the motorized operated valve program. The flow accelerated corrosion (FAC) program will be affected (i.e., new values for flow, pressure and temperature). However, the identified changes are not significant and are not expected to cause inspection intervals or repairs to increase significantly following the 1.4 percent MUR power uprate.

3.4.6 NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"

NRC Bulletin 88-02, dated February 5, 1988, requested holders of Westinghouse designed nuclear power reactors with steam generators (SGs) having carbon steel support plates to implement actions to minimize the potential for a steam generator tube rupture. Point Beach responded to the Bulletin stating that it only applied to Unit 2. The Unit 1 SGs had been replaced in 1983 with a design that included tube support plates designed to minimize the potential for denting of the tubes. The Unit 1 support plates are made of SA-240 Type 405 stainless steel and have quatrefoil shaped holes. Therefore, Bulletin 88-02 does not apply to PB Unit 1. Actions were put in place for the Unit 2 SGs since the tube support plates were of carbon steel. In 1996, the Unit 2 SGs were replaced. The replacement steam generators for Unit 2 also have stainless steel support plates with trifoil broached tube holes. This design minimizes the potential for denting in the Unit 2 SGs.

In conclusion, the NRC Bulletin 88-02 does not apply to the PBNP Unit 1 and Unit 2 SGs because both have tube support plates that are designed to minimize the potential for this type of corrosion. The MUR power uprate does not change this. Additionally, steam generator inspections are performed in accordance with PBNP NP 7.7.16, "Steam Generator Program," and NP 7.7.17, "Requirements for Steam Generator Primary Side Activities." These procedures incorporate the requirements of NEI 97-06, "Steam Generator Program Guidelines." Any degradation due to denting would be identified and evaluated through this program.

3.4.7 References for Section 3.4

3.4.7.1 ASME Boiler and Pressure Vessel Code, Section III, *Nuclear Vessels*, 1965 Edition, American Society of Mechanical Engineers, New York.

3.4.7.2 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1968 Edition with addenda through Winter 1968, American Society of Mechanical Engineers, New York.

3.4.7.3 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1974 Edition, American Society of Mechanical Engineers, New York.

3.4.7.4 ASME Boiler and Pressure Vessel Code, Section III-NB, Summer 1966 though Winter 1969 Addenda, American Society of Mechanical Engineers, New York.

3.4.7.5 USA Standards B31.1, Power Piping Code, 1967.

3.4.7.6 ASME Boiler and Pressure Vessel Code, Section III, *Rules for Construction of Nuclear Power Plant Components*, 1986 Edition, American Society of Mechanical Engineers, New York.

3.4.7.7 Westinghouse Equipment Specification 676433, Rev. 1, "PBNP Station 1 and 2 Reactor Coolant Controlled Leakage Pump," RC Moren, October 9, 1967.

3.4.7.8 ASME Boiler and Pressure Vessel Code, Section III, *Rules for Construction of Nuclear Vessels*, 1965 Edition with Addenda through Summer 1966, American Society of Mechanical Engineers, New York.

3.4.7.9 PBNP FSAR Appendix A.5, "Seismic Design Analysis," 06/01.

3.4.7.10 AISC, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, eighth edition.

3.4.7.11 ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, *Component Supports*, 1974 Edition, American Society of Mechanical Engineers, New York.

3.4.7.12 ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and Appendices, 1968 Issue and summer 1968, American Society of Mechanical Engineers, New York.

3.4.7.13 ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 1989, American Society of Mechanical Engineers, New York.

3.4.7.14 ASME Boiler and Pressure Vessel Code, Section III, 1977 through Winter 1978 Addenda.

3.4.7.15 WCAP-14602, Volume 1, PBNP Unit 2 Replacement Steam Generator Engineering Report, March 1996.

3.4.7.16 WEP-01-226, PCWG-2699, "NSSS Design Parameters for 1.4 Percent Mini Uprate," December 7, 2001.

3.4.7.17 WCAP 14459, "Reactor Pressure Vessel and Internals System Evaluations for the Point Beach Units 1 and 2 Power Uprating/Replacement Steam Generator," 1996.

3.4.7.18 "Reload Transition Safety Report Point Beach Units 1 and 2 422Vantage+ Fuel Upgrade/Power Uprate," Westinghouse Report, Rev. 2, November, 1999.

3.4.7.19 NPL 99-0369, "TS Change Request 210, Amendment to facility operating Licenses to Reflect Required Changes to the Technical Specifications as a Result of Using Upgraded Fuel," 06/22/99.

3.5 Electrical Equipment Design

3.5.1 Normal Electrical Power Systems

The MUR power uprate was evaluated for impact on the main power transformers, main generators, unit auxiliary transformers (UATs), low voltage station auxiliary transformers (LSATs), and the 4160 volt buses. The evaluation indicated that the increased generator loads (approximately 7.5 MWe) and the increased 4160V loads (approximately 0.08 MVA per bus) had minimal impact on the available margins for each of these systems.

3.5.2 Emergency Diesel Generators

The MUR power uprate was evaluated for impact on the Emergency Diesel Generators (EDG). No equipment changes as a result of the MUR power uprate were identified that would cause the loads on the emergency diesel generators to change. Therefore, it is concluded that the EDGs are adequate for the MUR uprate.

3.5.3 Station Blackout Equipment

The only potential impact of the MUR power uprate on the ability of the plant to withstand and recover from a station blackout (SBO) is the increased decay heat that must be removed from the RCS. The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pumps, supplied with steam from the steam generators, to support reactor heat removal due to uprate. The Technical Specification minimum required volume in the CST is 13,000 gallons. This volume remains acceptable for the 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 PBNP to respond to a SBO will not be altered due to the MUR power uprate.

3.5.4 Environmental Qualification of Electrical Equipment

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive postulated harsh environments during normal operation and post-accident. This includes conditions of normal operation and design basis events (e.g., LOCA). The environmental conditions for equipment qualification requirements are defined for the PBNP in Design Guideline (DG-G11). This design guideline summarizes the environmental conditions resulting from normal, abnormal, and accident operating conditions. Included environmental conditions are: Pressure, Temperature, Relative Humidity, Radiation, Chemical Spray, and Submergence.

PBNP EQ conditions or parameters are established for normal and accident operating conditions in the containment, and portions of the auxiliary and turbine buildings. Changes to these operating conditions were evaluated for a 1678 MWt extended power uprate. The evaluations encompass the 1.4 percent MUR power uprate. The power uprate does not affect the chemical spray, submergence, or seismic aspects of the PBNP design and, therefore, these parameters are not discussed below.

Normal Operating Conditions

Pressure / Temperature / Humidity

Normal service conditions are those environments that are maintained in each area during normal plant operation. The power uprate results in changes to the RCS, steam generators, main steam, and feedwater parameters. An uprate as high as 10.5 percent corresponding to 1678 MWt has been analyzed for the impact on the normal design temperatures and the environmental conditions in containment, the auxiliary building and the turbine building can be maintained within the normal ranges by the existing HVAC. Therefore, the normal aging conditions in these areas will not be impacted and are bounded by the current design.

Radiation

The effects of the power uprate on the radiation environments are discussed in Section 3.3. Please refer to Section 3.3 for a detailed summary.

Accident Conditions

Accident conditions are the most severe environments that may occur in each area following a postulated accident. In general, accidents causing the most severe environments include loss-of-coolant accidents and main steam line breaks inside containment and high energy line breaks outside containment.

Pressure / Temperature / Humidity
The main steam system operating conditions are lower for the MUR power uprate, and the design pressure and temperature remain unchanged. The main steam line break results for inside (pending approval, Reference 3.5.6.1) containment will bound the conditions for the MUR uprate since the analyses are performed at 1549 MWt (102 percent current power). Mass and energy releases for a steam line break outside containment were taken from a Westinghouse generic source and remain conservative for MUR uprated conditions. The resulting pressures, temperatures, and humidity values calculated for steam line break outside of containment do not change and remain bounding for the MUR uprate. Therefore, there are no changes to these parameters for a steam line break accident.

The LOCA containment response is currently analyzed at 1549 MWt (102 percent of current power). Therefore, the PBNP containment analysis does not change for the MUR uprate and there are no changes in temperature, pressure, and humidity following a LOCA.

Radiation
The effects of the power uprate on the radiation environments are discussed in Section 3.3. Please refer to Section 3.3 for a detailed summary and 3.6.6 for Control Room Ventilation/Habitability.

3.5.5 Grid Stability

American Transmission Company (ATC) assessed grid stability and thermal loading for the Point Beach Nuclear Plant Units 1 and 2. The impact study performed by ATC identified no stability issues for the 1.4 percent MUR uprate for the PBNP units.

3.5.6 References for Section 3.5

3.5.6.1 NMC Letter, NRC 2002-0004, "License Amendment Request 223 Containment Pressure," 01/11/2002.

3.6 System Design

3.6.1 NSSS/BOP Interface Systems

Main Steam System

Main Steam Safety Valve (MSSV) capacity is dictated by safety analysis. The results of the current accident analysis at 1650 MWt conclude that the installed safety valves have adequate capacity. This analysis will remain bounding for the smaller, 1.4 percent MUR power uprate to 1540 MWt.

The capacity of the atmospheric steam dump valves (ASDVs) also conclude that the original design basis in terms of cooldown capability can still be achieved for the small MUR power uprate.

The NSSS/BOP interface systems requirement imposed on the design of the Main Steam Isolation Valves (MSIVs), not-return check valves and associated pipe loads are not impacted by the power uprate. Additionally, the condenser dump valve capacity remains adequate to handle load rejections up to 50 percent of full power as indicated in the current PBNP FSAR.

Condensate and Feedwater

The impact of the MUR on the condensate and feedwater systems was evaluated using heat balances derived from plant specific PEPSE models and pump curves for the condensate pumps, main feedwater pumps, and heater drain pumps. The evaluation indicated that impact is minimal with flow through each of the systems increasing 2 percent or less. This increase in flow is well within the design margin of the systems. In addition, the increased feedwater flow is expected to result in a 2 percent or less change from the current main feedwater regulating valve position (approximately 75 percent).

Auxiliary Feedwater System (AFW)

The minimum AFW flow requirements are dictated by the accident analysis. The loss of normal feedwater (LONF) event is the most limiting with respect to the required flow capacity of the AFW system. The current accident analyses were performed at 1549 MWt (initial power) using the current flow capacity of the AFW motor driven pumps. The turbine driven AFW pumps have even more capacity and are, therefore, also able to meet system requirements. Therefore, no changes are necessary for the MUR power uprate.

The AFW pumps take suction from the Condensate Storage Tanks (CST). The limiting transient for CST inventory requirement is the Station Blackout (SBO) event. The current minimum CST inventory requirement in Technical Specifications is based on the SBO coping period and the ability to support decay heat removal. The Technical Specification CST volume of 13,000 gallons is based on a calculation that assumed an initial core power of 1587 MWt. This core power is greater than the 1540 MWt RTP requested for the MUR power uprate. Therefore, the current minimum CST inventory requirement is acceptable for supporting the decay heat at the MUR uprated core power of 1540 MWt. No changes are necessary for the MUR power uprate.

Steam Generator Blowdown

The blowdown flow rate required to control chemistry and the buildup of solids in the steam generator is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water, and allowable primary to secondary leakage. These variables are not impacted by the power uprate and, therefore, the blowdown required to control the secondary chemistry and steam generator solids will not be impacted by the power uprate.

The inlet pressure to the SG blowdown system varies with the SG operating pressure. The present range of approved NSSS design parameters permits a range of operating pressures that bound the power uprate design parameters. The MUR uprated parameters will remain below the system and component design conditions.

3.6.2 Containment Systems

Containment System Structure and Containment Isolation System

No changes to the containment structure or containment isolation systems are being made as part of the MUR power uprate. The systems are periodically tested for containment design integrity. There are no changes in the test programs based on the MUR uprate. The Steam Line Break (SLB) (pending approval, Reference 3.6.7.1) and Loss of Coolant Accident (LOCA) containment integrity analyses are based on 102 percent of the current licensed power (1518.5 MWt). Both analyses bound operation at the MUR uprated power of 1540 MWt. Therefore, the MUR uprate does not affect these systems.

Containment Ventilation System

The containment ventilation system is a safety-related system designed to remove heat from containment following a LOCA or a main steam line break (MSLB) inside containment. The system limits the containment temperatures and pressures to less than the containment design limits. The system also has the non-safety related function of removing heat from containment during normal plant operation.

The containment ventilation system was evaluated for normal and accident operations to determine if the system would need modification as a result of additional heat loads from a full 10.5 percent power uprate to 1678 MWt. A conservative increase of two percent in heat load during normal operation was assumed in the analysis for an extended uprate. The two percent increase in heat load would not adversely impact the system operation. It was determined that for accident conditions, the containment fan coil units and the containment spray system could still mitigate the increase in energy released from a LOCA. The MUR power uprate would be bounded by this evaluation since the ventilation heat load increase would be much smaller. Therefore, the containment ventilation system remains capable of performing its functions following the MUR power uprate.

3.6.3 Safety-related cooling water systems

Safety Injection (SI)/Containment Spray (CS) System

The existing SI and CS systems have been evaluated for an uprated power of 1678 MWt. There are no changes to these systems or the current containment integrity analysis performed at 1549 MWt. Therefore, the systems remain acceptable for the MUR power uprate.

Residual Heat Removal (RHR) System

The existing RHR System capability has been assessed and found to be adequate for an uprated power condition of 1549 MWt. Both the Appendix R and Normal Cooldown requirements for the system are satisfied at the MUR power uprate condition.

Service Water (SW) System

The PBNP SW system is the major cooling system for safety-related and non-safety-related equipment. The MUR power uprate will increase the heat rejection to the service water system slightly. This very slight increase will not cause system design pressures and temperatures to be exceeded and the system will continue to satisfy its normal and accident function without any modifications to the system or means of operating the system. Furthermore, the SW system was reviewed for an extended 10.5 percent power uprate to 1678 MWt and similar conclusions were found.

Component Cooling Water System (CCWS)

The CCWS is adequately sized for normal cooldown heat loads associated with a power uprate of 1549 MWt. It is also adequately sized to meet the Appendix R cooldown requirements. Small changes in heat loads that are predicted to occur during normal modes of plant operation are well within the system's design capability. Therefore, the CCWS requirements are met for a 1549 MWt uprate.

3.6.4 Spent Fuel Pool (SFP) Storage and Cooling Systems

The existing Spent Fuel (SF) Pool Cooling System capability has been assessed and found to be adequate for an uprated core power of 1678 MWt. Performance requirements related to both safety and non-safety related functions of the SF System have been reviewed for impact from power uprate conditions. This assessment was based on conservative assumptions that included (a) an instantaneous core offload of fuel into the SFP, and (b) that the SFP contains the maximum capacity of fuel assemblies (1502). The system can maintain the maximum normal operating temperature of 145°F, assuming the "instantaneous" minimum core off load time of approximately 100 hours (60°F service water) and 270 hours (80°F service water) after shutdown. Additionally, prior to each refueling, the decay heat generation rate from the fuel inventory in the SFP is calculated to ensure that it is within the limits of the capability of the SF System.

3.6.5 Radwaste Systems

The PBNP FSAR Sections 11.1 and 11.2 detail the Waste Liquid System (WL) and the Gaseous Waste Management System (WG), respectively. The design basis of the WL system is to provide retention of liquid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified on the basis of 10 CFR 20 requirements, for both normal operations and any transient situation that might reasonably be anticipated to occur, and on the basis of 10 CFR 100 dose level guidelines for potential reactor accidents of exceedingly low probability of occurrence. The design basis for the WG system is the same except that it is specifically for the retention of gaseous effluents.

The WL and WG systems have been previously evaluated and estimates of annual releases are based on a core power of 1683 MWt. This power level bounds that of the smaller MUR power uprate. Additionally, no equipment is changing or being added to these systems. Therefore, the system evaluations discussed in the FSAR for the WL and WG systems remain applicable.

Section 3.3 also evaluated the effect of the uprate on normal effluents and the original 10 CFR 50, Appendix I analysis.

3.6.6 Engineered Safety Feature (ESF) Ventilation Systems

An evaluation of the Heating, Ventilation, and Air Conditioning (HVAC) systems were performed for a 10.5 percent uprate to 1678 MWt. The scope of the evaluation included the Auxiliary Building and Control Building HVAC, the Containment Heating and Ventilation System, the Control Room HVAC System, and other HVAC systems that support plant operation. This thermal uprate (to 1678 MWt) results in an increased heat loss to the environments that house the main steam, steam generator blowdown and feedwater piping. Other areas may also experience slight increases in heat load. These slight increases in heat loads do not affect the ability of the HVAC systems to operate satisfactorily following uprate. Additionally, an uprate to 1678 MWt would not affect the HVAC systems' abilities to perform non-cooling functions (i.e., isolating containment, maintaining negative pressure, filtering particulates or iodine, heating or providing ventilation to reduce hydrogen concentration). It should be noted that with the exception of the Control Room HVAC, PBNP does rely on any cleanup filtration systems post-accident.

Therefore, the function of the PBNP ventilation systems is not affected and the systems' ability to maintain operating temperatures within acceptance limits is not impacted by an uprate to 1678 MWt. Additionally, the MUR uprate would result in an even smaller change in environmental conditions or in design margins. Therefore, the MUR power uprate is bounded by the evaluation completed for the uprate to 1678 MWt.

Containment ventilation, and its safety related function, is described in section 3.6.2. Control Room Ventilation and Habitability are discussed below.

Control Room Ventilation/Habitability

The PBNP control room ventilation system is a non-safety related system that provides heating, ventilation, air conditioning and radiological habitability for the control room envelope. An uprate to 1678 MWt will not significantly change the heat loading of the control room under any normal or accident condition. Therefore, the increased heat loads associated with the smaller MUR power uprate to 1540 MWt, are bounded and the system will continue to operate acceptably.

The LOCA radiological analysis is the limiting analysis for post-accident control room operator dose. The current LOCA radiological analysis was performed to support the Unit 2 Steam Generator Replacement Program and issues with post-accident cooling capabilities. NRC SERs dated July 1 and July 9, 1997, approve the LOCA radiological analysis and state that it remains the bounding analysis for control room operator dose. The LOCA radiological analysis was performed at 1549 MWt (102 percent of 1518.5 MWt) and encompasses the MUR power uprate. Additionally, there are no changes to the control room ventilation system and the heat loads.

SFP Area Ventilation Systems

As indicated above, the slight increases in heat loads do not affect the ability of the HVAC systems to operate satisfactorily following uprate. The spent fuel pool area ventilation falls under two PBNP HVAC systems: The Primary Auxiliary Building Heating and Ventilating (VNPAB) and the Drumming Area Heating and Ventilating (VNDRM). These systems are non-safety related and are not relied upon in mitigating offsite doses following a fuel handling accident. The minimal heat loads added to the PAB due to the 1678 MWt uprate will not impact the VNPAB system's performance. The only thermally related function of the VNDRM is to heat outside air during the winter. Its other function is to limit radiation doses to personnel from drumming operations or from radioactive vapor. The operation of the VNDRM is not affected by the uprate to 1678 MWt. The MUR power uprate is bounded by the evaluation for the 1678 MWt uprate.

3.6.7 References for Section 3.6

3.6.7.1 NMC Letter, NRC 2002-0004, "License Amendment Request 223 Containment Pressure," 01/11/2002.

3.7 Other Evaluations

3.7.1 Operator Actions Sensitive to a 1.4 Percent MUR Power Uprate

The MUR power uprate is not expected to have any significant effect on the manner in which the operators control the plant (including operator response times) during normal operations or transient conditions. All operator actions that were taken credit for in the safety analysis are still valid following the MUR power uprate since the current safety analyses are all performed at either 1549 MWt or at 1650 MWt. Additionally, PBNP has performed an evaluation to determine the impact of the MUR power uprate on the PBNP Probabilistic Safety Assessment (PSA) model. This included a review of the Human Reliability Analysis PRA Notebook. It was concluded that the MUR power uprate would cause no changes in the timing for operator actions assumed in the PSA.

3.7.2 Affects of MUR Uprate Modifications on Plant Operations

Plant operation has been reviewed to determine the affects of the MUR power uprate plant modifications. These plant modifications include the installation of the LEFM electronics unit, the replacement of the transducers on each unit's spool piece, and software changes to the Plant Process Computer System (PPCS). The following aspects of plant operation were reviewed to determine the affects of the MUR power uprate modifications: emergency and abnormal operating procedures; control room controls, alarms and displays; simulator; and operator training program. Each is evaluated below with accompanying statements of the measures taken to ensure that the changes in operator action do not adversely affect defense in depth or safety margins. The measures or changes identified below will be completed prior to the implementation of the MUR uprate license amendment (i.e., increasing RTP to 1540 MWt).

Emergency and Abnormal Operation Procedures

There currently are no Emergency Operating Procedures (EOP) that reference use of the LEFM. Additionally, use of the LEFM will not be incorporated into any Emergency Operating Procedures. Abnormal Operating Procedure, AOP-21, "PPCS Malfunction," is referred to when either the PPCS or certain PPCS monitoring functions become unavailable. A revision to this procedure will direct operators to the proposed TRM Section 3.3.2 describing the out of service actions and the TRM Limiting Conditions for Operation (TLCO) for an inoperable LEFM. This procedure, as well as normal operating procedures, will be changed to reference the TRM as appropriate for LEFM operability. Any changes associated with the power uprate will be treated in a manner consistent with the site design modification process and will be included in Operator Training accordingly.

Control Room Controls, Displays and Alarms

The installation and use of the LEFM adds 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 will not change. There are no new controls for the operator to manipulate. The Alarm Response Book (ARB) will be updated accordingly. The reactor operators will 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.

Control Room Plant Reference Simulator

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

Operator Training Program

The installation of the LEFM and implementation of the MUR power uprate license amendment (i.e., the increase in RTP) will require procedure and training changes. Actions will 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 will be completed prior to implementation of the power ascension to 1540 MWt. All of this information will be updated in a manner consistent with other plant modifications and license amendments.

3.7.3 Environmental Evaluation

3.7.3.1 Changes to the Types or Amounts of Any Effluents

A review considering the operating license, the current Wisconsin Pollutant Discharge Elimination System Permit (WPDES) permits, and the information contained in the Final Environmental Statement (FES) was performed. Effluents from the plant that could change as a result of the MUR uprate are thermal discharges to Lake Michigan and radiological effluents. Although increases in discharge amounts associated with the proposed thermal power uprate are possible, they will remain within acceptable limits. Annual discharges will continue to be a small percentage of the allowable limits and the FES estimates. The effluents are described below.

Changes to the circulating water system operating parameters were evaluated with regard to the temperature limits on discharge water for an uprate of 10.5 percent. The uprated power of 1678 MWt resulted in a circulating water increase of 2°F. The temperature increase associated with the MUR uprate to 1540 MWt would be less. Additionally, the PBNP Wisconsin Pollutant Discharge Elimination System Permit, WI-0000957-6, does not limit maximum discharge temperature, delta T across the condenser, or total discharge heat. Therefore, this permit does not require modification as a result of the MUR uprate.

Annual radiological effluents were evaluated for an uprate to 1678 MWt. These effluents were described in section 3.3 of this report. Based on the evaluations performed for an uprated power of 1683 MWt the liquid and gaseous radwaste system design will be capable of maintaining normal operational offsite releases and doses within the requirements of 10 CFR 20 and 10 CFR 50, Appendix I. Additionally, effluent increases are assumed to be proportional to the increase in power. Therefore, effluents from the MUR power uprate (1540 MWt) are bounded by this evaluation. Solid waste volume generation is expected to increase slightly. However, all solid waste is controlled within several state and federal regulatory limits through the PBNP Process Control Program.

3.7.3.2 Individual or Cumulative Occupational Exposure

The MUR power uprate will impact the radiation source terms in the core and the coolant. Normal operation radiation levels are expected increase approximately 1.4 percent (i.e., the percentage of the core uprate). The actual increase in radiation levels due to uprate will not significantly affect radiation zoning in the various areas of the plant for the following reasons: 1) the increase in uprated radiation levels will be offset by the conservatism assumed in the original design basis source terms that were used to establish the radiation zones, 2) plant Technical Specifications limit the RCS concentration levels well below the design basis source terms, and 3) conservative analytical techniques are used to establish shielding requirements. Regardless of the above, individual worker exposures will be maintained within the acceptable limits of the site ALARA program that controls access to radiation areas.

3.7.3.3 Environmental Review Conclusions

Thermal discharges and radiological effluents may change slightly following the MUR uprate. However, these changes have been evaluated and the changes remain within the regulatory limits or permit limits. Radiation exposure was also reviewed. Although dose rates will increase proportionally with the percent of uprate, the doses will remain within the regulatory limits. The site ALARA program will continue to monitor and control personnel exposure such that the regulatory limits are not exceeded.

Based on the above, the proposed change does not involve a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed change is not required.

3.8 Technical Specification, Protection System Settings, Emergency Settings Changes

3.8.1 Technical Specification changes

Technical Specification changes are described in detail in section 2.0 of this report. Please refer to that section for this information.

3.8.2 Reactor Protection System Settings

A review of the Reactor Protection System settings of TS 3.3.1 was performed. It was noted that several of the reactor protection system functions of TS Table 3.3.1-1 could potentially be affected as a result of the MUR power uprate. Those are listed below by function. None of the functions require a Technical Specification change.

Function 2, "Power Range Neutron Flux," both High and Low, will change as a result of power uprate. The change does not require a TS change but rather a change in current scaling for the 100 percent power value. Similarly, Functions 3 and 4, "Intermediate Range Neutron Flux," and "Source Range Neutron Flux," will also be changing, but will not require a TS change.

Functions 5 and 6, Overtemperature ΔT and Overpressure ΔT both have the ΔT_0 (ΔT at rated power) changing in the calculation. This does not require a TS change, but requires changes to the procedure calibrating ΔT_0. It also requires changes to the COLR.

Function 14, "Steam flow/Feedwater Flow Mismatch," will not require change since the allowable value looks at a difference between the two flow measurements. The value of the difference is not affected by the MUR uprate.

All of the above changes will occur with the implementation of this proposed amendment. The changes will be performed using appropriate plant modification and procedure change processes.

3.8.3 Emergency Settings

Two Engineered Safety Feature Actuation System (ESFAS) Instrumentation functions could potentially be affected as a result of the MUR power uprate. TS Table 3.3.2-1, Function 4, "Steam Line Isolation," d., "High Steam Flow," currently states an allowable value of 0.66 E 06 lb/hr at 1005 psig. This value will increase by 1.4 percent for the MUR power uprate. However, a 1.4 percent increase results in a very small increase in the allowable value to 0.669 E 06 lb/hr. It is conservative to leave the trip set point at the pre-uprate value since it would cause an earlier trip. The margin to spurious trips is judged to remain adequate. Therefore, this ESFAS allowable value will not change for the MUR power uprate.

The second ESFAS function is the allowable value for TS Table 3.3.2-1, Function 4, "Steam Line Isolation," e., "High High Steam Flow." This value is currently based on 120 percent of full steam flow at full steam pressure. The allowable value corresponds to 4 E 06 lb/hr at 806 psig. The value will also change proportional with the power uprate. Therefore, the increased allowable value would be 4.06 E 06 lb/hr. Again, it is conservative to trip early, and therefore, this allowable value can remain at the value currently stated in TS Table 3.3.2-1.

The ESFAS functions described above do not require change for implementation of the MUR power uprate because the change is very small and leaving the trip set point values the same is conservative (the plant would trip earlier).

4.0 REGULATORY ANALYSIS

4.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company (licensee) hereby requests amendments to facility operating licenses DPR-24 and DPR-27, for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to revise the operating licenses and the Technical Specifications to allow operation at an increased rated thermal power (RTP) of 1540 MWt.

Nuclear Management Company has evaluated the proposed amendments in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the Point Beach Nuclear Plant in accordance with the proposed amendments presents no significant hazards. Our evaluation against each of the criteria in 10 CFR 50.92 follows.

1. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

The comprehensive analytical efforts performed to support the proposed change included a review of the FSAR Chapter 14 Accident Analysis, the Nuclear Steam Supply System (NSSS) systems and components, Electrical Equipment, and Balance of Plant Systems. There are no changes as a result of the MUR power uprate to the design or operation of the plant that could affect system, component or accident mitigative functions. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

The reduction in power measurement uncertainty allows for most of the safety analyses to continue to be used without modification. This is because the safety analyses were performed or evaluated at either 1650 MWt or 102 percent of 1518.5 MWt. This supports a core power level of 1540 MWt with a measurement uncertainty of 0.6 percent. Radiological consequences of Chapter 14 accidents were assessed previously using uprated cores and continue to be bounding. The FSAR Chapter 14 analyses continue to demonstrate compliance with the relevant accident analyses acceptance criteria. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms, loop piping and supports, reactor coolant pump, steam generators, and pressurizer) were evaluated at 1650 MWt and continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no significant increase in the probability of a structural failure of these components.

All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/Balance of Plant (BOP) interface systems were evaluated and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. No equipment modifications to these systems are planned for this change. Therefore, there is no significant increase in the probability of an accident previously evaluated.

2. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.**

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety-related systems or component and does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.**

Operation at the 1540 MWt core power does not involve a significant reduction in the margin of safety. Most of the current accident analyses and system and component analyses had been previously performed at uprated core powers that exceed the MUR uprated core power. Evaluations have been performed for analyses that were done at nominal core power and have been found acceptable for the MUR power uprate. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been either reviewed and approved by the NRC or are in compliance with applicable regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Conclusion

Operation of the PBNP in accordance with the proposed amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and does not result in a significant reduction in a margin of safety. Therefore, operation of PBNP in accordance with the proposed amendments does not involve a significant hazards consideration.

ATTACHMENT 4

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 226</u>

Proposed Facility Operating License And Technical Specification Changes (marked up)

A. Maximum Power Levels

NMC is authorized to operate the facility at reactor core power levels not in excess of 1540~~18.5~~ megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. __~~202~~,- are hereby incorporated in the license. NMC shall operate the facility in accordance with Technical Specifications.

C. Deleted

D. Deleted

E. Spent Fuel Pool Modification

The licensee* is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

F. NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FFR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Point Beach Nuclear Plant Modified Amended Security Plan," with revisions submitted through March 23, 1988; "Point Beach Nuclear Plant Modified Amended Security Force Training and Qualification Plan," with revisions submitted through August 6, 1982; and "Point Beach Nuclear Plant Modified Amended Security Contingency Plan," with revisions submitted through March 6, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

* Reference to the licensee in License Conditions 3.E, 3.G and 3.J refers to Wisconsin Electric Power Company and is maintained for historical purposes.

1. This amended license applies to the Point Beach Nuclear Plant Unit No. 2, a closed cycle, pressurized, light water moderated and cooled reactor, and associated steam generators and electric generating equipment (the facility). The facility is located on the Point Beach site, in the Town of Two Creeks, Manitowoc County, Wisconsin, and is described in the "Final Safety Analysis Report", as supplemented and amended.

2. Subject to the conditions and requirements incorporated here in the Commission hereby licenses

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Wisconsin Electric Power Company to possess, and NMC to use and operate the facility at the designated location on the Point Beach site in accordance with the procedures and limitations set forth in this license;

   B. Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in Final Facility Description and Safety Analysis Report, as supplemented and amended as of March 17, 1976;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed source for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

   D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

3. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Levels

   NMC is authorized to operate the facility at reactor core power levels not in excess of 15~~18.5~~40 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ~~208~~___, are hereby incorporated in the license. NMC shall operate the facility in accordance with Technical Specifications.

C. Deleted

D. Deleted

E. Spent Fuel Pool Modification

The licensee* is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

F. Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Point Beach Nuclear Plant Modified Amended Security Plan," with revisions submitted through March 23, 1988; "Point Beach Nuclear Plant Modified Amended Security Force Training and Qualification Plan," with revisions submitted through August 6, 1982; and "Point Beach Nuclear Plant Modified Amended Security Contingency Plan," with revisions submitted through March 6, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

* Reference to the licensee in License Conditions 3.E and 3.G refers to Wisconsin Electric Power Company and is maintained for historical purposes.

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 15~~18.5~~40 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation; b. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and c. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(8) LCO 3.2.3, "Axial Flux Difference (AFD)"
(9) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11) LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12) LCO 3.9.1, "Boron Concentration"

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:~~specifically those described in the following documents:~~

(1) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)
(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
(3) WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(4) WCAP-14787-P, Rev. 1, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header), February, 2002.

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

(11) Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

(12) Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System," Revision 0, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ATTACHMENT 5

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

License Amendment Request 226

Proposed Facility Operating License and Technical Specification Changes (clean copy)

A. Maximum Power Levels

NMC is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. NMC shall operate the facility in accordance with Technical Specifications.

C. Deleted

D. Deleted

E. Spent Fuel Pool Modification

The licensee* is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

F. NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FFR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Point Beach Nuclear Plant Modified Amended Security Plan," with revisions submitted through March 23, 1988; "Point Beach Nuclear Plant Modified Amended Security Force Training and Qualification Plan," with revisions submitted through August 6, 1982; and "Point Beach Nuclear Plant Modified Amended Security Contingency Plan," with revisions submitted through March 6, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

* Reference to the licensee in License Conditions 3.E, 3.G and 3.J refers to Wisconsin Electric Power Company and is maintained for historical purposes.

1. This amended license applies to the Point Beach Nuclear Plant Unit No. 2, a closed cycle, pressurized, light water moderated and cooled reactor, and associated steam generators and electric generating equipment (the facility). The facility is located on the Point Beach site, in the Town of Two Creeks, Manitowoc County, Wisconsin, and is described in the "Final Safety Analysis Report", as supplemented and amended.

2. Subject to the conditions and requirements incorporated here in the Commission hereby licenses

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," Wisconsin Electric Power Company to possess, and NMC to use and operate the facility at the designated location on the Point Beach site in accordance with the procedures and limitations set forth in this license;

   B. Pursuant to the Act and 10 CFR Part 70, NMC to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in Final Facility Description and Safety Analysis Report, as supplemented and amended as of March 17, 1976;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed source for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

   D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NMC to receive, possess and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, NMC to possess such byproduct and special nuclear materials as may be produced by the operation of the facility, but not to separate such materials retained within the fuel cladding.

3. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Levels

      NMC is authorized to operate the facility at reactor core power levels not in excess of 1540 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ___, are hereby incorporated in the license. NMC shall operate the facility in accordance with Technical Specifications.

C. Deleted

D. Deleted

E. Spent Fuel Pool Modification

The licensee* is authorized to modify the spent fuel storage pool to increase its storage capacity from 351 to 1502 assemblies as described in licensee's application dated March 21, 1978, as supplemented and amended. In the event that the on-site verification check for poison material in the poison assemblies discloses any missing boron plates, the NRC shall be notified and an on-site test on every poison assembly shall be performed.

F. Physical Protection

NMC shall fully implement and maintain in effect all provisions of the Commission-approved physical security guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Point Beach Nuclear Plant Modified Amended Security Plan," with revisions submitted through March 23, 1988; "Point Beach Nuclear Plant Modified Amended Security Force Training and Qualification Plan," with revisions submitted through August 6, 1982; and "Point Beach Nuclear Plant Modified Amended Security Contingency Plan," with revisions submitted through March 6, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

* Reference to the licensee in License Conditions 3.E and 3.G refers to Wisconsin Electric Power Company and is maintained for historical purposes.

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1540 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation; b. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and c. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(8) LCO 3.2.3, "Axial Flux Difference (AFD)"
(9) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overtemperature ΔT"
(10) LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation - Overpower ΔT"
(11) LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
(12) LCO 3.9.1, "Boron Concentration"

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:

(1) WCAP-14449-P-A, "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWR's with Upper Plenum Injection," Revision 1, October 1999. (cores containing 422V+ fuel)
(2) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
(3) WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

(4) WCAP-14787-P, Rev. 1, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header)," February, 2002.

(5) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using The NOTRUMP Code," August 1985.

(6) WCAP-10054-P-A, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Addendum 2, Revision 1, July 1997.

(7) WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

(8) WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control," Revision 1A, February 1994.

(9) WCAP-10924-P-A, "Large Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addenda, December 1988. (cores not containing 422 V+ fuel)

(10) WCAP-10924-P-A, "LBLOCA Best Estimate Methodology: Model Description and Validation: Model Revisions," Volume 1, Addendum 4, August 1990. (cores not containing 422 V+ fuel)

(11) Caldon, Inc., Engineering Report-80P, "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

(12) Caldon, Inc., Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System," Revision 0, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP enabling, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(1) LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
(2) LCO 3.4.6, "RCS Loops-MODE 4"
(3) LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled"
(4) LCO 3.4.10, "Pressurizer Safety Valves"
(5) LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC Letters dated October 6, 2000 and July 23, 2001.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in a report for the period in which the inspection was completed.

Reports shall include:

1. Number and extent of tubes inspected.
2. Location and percent of all thickness penetration for each indication.
3. Identification of tubes plugged or repaired.

(c) Reports required by Table 5.5.8-1, "Steam Generator Tube Inspection," shall provide the information required by Specification 5.6.8.(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

ATTACHMENT 6

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 226</u>

List of Regulatory Commitments

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	Due Date/Event
1. PBNP will complete revisions to affected documents (i.e., procedures, TRM, and Alarm Response Book) and provide appropriate training to the necessary plant staff for changes associated with the installation of the LEFM and the implementation of the uprated RTP (Attachment 1, Section 3.1.4, 3.1.5, 3.7.2).	1. Prior to MUR uprate implementation.
2. PBNP will document a formal engineering evaluation of FSAR Chapter 14.2.3, "Accidental Release – Waste Gas," at uprate conditions (Attachment 1, Section 3.3.1).	2. Prior to MUR uprate implementation.
3. PBNP will complete the qualification of butyl rubber insulated cable and will update other EQ documentation to address the increased radiation levels due to the uprate conditions (Attachment 1, Section 3.3.2.3).	3. Prior to MUR uprate implementation.
4. PBNP will complete the resolution of the vital access issues prior to implementation of the uprate (Attachment 1, Section 3.3.2.4).	4. Prior to MUR uprate implementation.
5. PBNP will complete necessary changes to the Reactor Protection System Settings using appropriate plant modification and procedure change processes (Attachment 1, Section 3.8.2).	5. Prior to MUR uprate implementation.

ATTACHMENT 7

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

License Amendment Request 226

Westinghouse Proprietary Authorization Letter, CAW-02-1516

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

April 9, 2002

CAW-02-1516

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-14787, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company, Point Beach Units 1 and 2 (Fuel Upgrade and Uprate to 1656 MWt-NSSS Power with Feedwater Venturis, or 1679 MWt-NSSS Power with LEFM on Feedwater Header) (Proprietary)"

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the subject material is further identified in Affidavit CAW-02-1516 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1516 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: D. Holland/NRR

ATTACHMENT 8

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 226</u>

Westinghouse Affidavit

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
Before me this 10th day
of April, 2002

Lorraine M. Piplica
Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilized a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system, which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information, which is marketable in many ways. The extent to which such information is available to competitors diminished the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-14787, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology For Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 Mwt-NSSS Power with Feedwater Venturis, or 1679 Mwt-NSSS Power with LEFM on Feedwater Header", (Proprietary), January 2002, being transmitted by Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by Nuclear Management Company for Point Beach Units 1 & 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of use of the Revised Thermal Design Procedure Instrument Uncertainty Methodology.

This information is part of that which will enable Westinghouse to:

(a) Provide power calorimetric uncertainty with Leading Edge Flow Meter on Feedwater Header.

(b) Establish appropriate procedures for calculation of calorimetric uncertainty with Leading Edge Flow Meter on Feedwater Header.

(c) Assist the customer to obtain NRC approval.

Further, this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(d) Westinghouse can sell support and defense of the calculation of power calorimetric uncertainty with Leading Edge Flow Meter on Feedwater Header.

ATTACHMENT 9

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

License Amendment Request 226

Westinghouse Proprietary Information Notice

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 OF THE Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

ATTACHMENT 10

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

License Amendment Request 226

Westinghouse Copyright Notice

COPYRIGHT NOTICE

ATTACHMENT 3

To

Letter from Mark Warner (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 226</u>

Non-Proprietary Version of WCAP-14788, Revision 1, Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company, Point Beach Units 1 & 2, (Fuel Upgrade & Uprate to 1656 MWt - NSSS Power with Feedwater Venturis, or 1679 MWt- NSSS Power with LEFM on the Feedwater Header)

Westinghouse Non-Proprietary Class 3

WCAP-14788
Revision 1

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Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 Mwt-NSSS Power with Feedwater Venturis, and 1679 Mwt-NSSS Power with LEFM on Feedwater Header)

Westinghouse Electric Company LLC

WCAP-14788
Rev. 1

Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Wisconsin Electric Power Company Point Beach Units 1 & 2 (Fuel Upgrade & Uprate to 1656 Mwt-NSSS Power with Feedwater Venturis, and 1679 Mwt-NSSS Power with LEFM on Feedwater Header)

March 2002

W.H. Moomau

WESTINGHOUSE ELECTRIC COMPANY LLC
4350 Northern Pike
Monroeville, Pennsylvania 15146-2886

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Westinghouse Revised Thermal Design Procedure

Instrument Uncertainty Methodology

(Fuel Upgrade & Uprate To 1656 Mwt - NSSS Power, and

1679 Mwt - NSSS Power With LEFM On Feedwater Header)

I. INTRODUCTION

Revision 0 of this report was completed to support an upgrade to the fuel product at uprated conditions of 1656 Mwt - NSSS power for both Units 1&2 at the Point Beach Nuclear Plant (PBNP). The fuel product to satisfy the intended requirements is the Westinghouse 14 X 14 PERFORMANCE + 422 fuel assembly. To utilize this new fuel assembly at the uprated conditions, a new accident analysis was required in addition to recalculating and revising the Instrument Uncertainty Methodology. Revision 0 of this report superseded "ITDP Instrument Uncertainty Report" dated June 7, 1984 (84WE*-G-044).

Revision 1 of this report is being completed to support a 1.4% uprate from the uprated condition of 1656 Mwt-NSSS power. This 1.4% uprate is possible due to the installation of an ultrasonic Leading Edge Flow Meter (LEFM) on the feedwater header. This LEFM will be used to perform the required daily calorimetric power measurement and continuous Reactor Thermal Output (RTO) calculation on the Plant Process Computer System (PPCS). The previous fuel analysis was performed at 2% above the rated full power value of 1656 Mwt-NSSS. This 2% accounted for the power measurement uncertainty. Therefore, the improved LEFM feedwater flow measurement will allow an additional power increase equivalent to the difference between the 2% and the reduced power measurement uncertainty obtained using the LEFM. The new power measurement uncertainty is included in this revision.

Four operating parameter uncertainties are used in the uncertainty analysis of the Revised Thermal Design Procedure (RTDP). These parameters are Pressurizer Pressure, Primary Coolant Temperature (T_{avg}), Reactor Power, and Reactor Coolant System Flow. They are frequently monitored and several are used for control purposes. Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least every 24 hours. RCS flow is monitored by the performance of a calorimetric flow measurement at the beginning of each cycle. The RCS Cold Leg loop flow indicators are evaluated against the calorimetric flow measurement. Pressurizer pressure is a controlled parameter and the uncertainty reflects the control system. T_{avg} is a controlled parameter via the temperature input to the rod control system, and the uncertainty reflects this control system. The RTDP[1] is used to predict the plant's DNBR design limit. The RTDP methodology considers the uncertainties in the system operating plant parameters, fuel fabrication and nuclear and thermal parameters and includes the use of various DNB correlations. Use of the RTDP methodology requires that variances in the plant operating parameters are justified. The purpose of the following evaluation is to define the specific Point Beach Units 1 & 2 Nuclear Plant instrument uncertainties for the four primary system operating parameters which are used to predict the plant safety analysis DNBR design limit via the RTDP, and to determine the starting points of certain plant parameters in some of the accident analyses.

Westinghouse has been involved with the development of several techniques to treat instrumentation uncertainties. An early version (for D. C. Cook 2 and Trojan) used the methodology outlined in WCAP-8567 "Improved Thermal Design Procedure," (2,3,4) which is based on the conservative assumption that the uncertainties can be described with uniform probability distributions. Another approach is based on the more realistic assumption that the uncertainties can be described with random, normal, two sided probability distributions.(5) This approach is used to substantiate the acceptability of the protection system setpoints for many Westinghouse plants, e.g., D. C. Cook 2(6), V. C. Summer, Wolf Creek, Millstone Unit 3 and others. The second approach is now utilized for the determination of all instrumentation uncertainties for the RTDP parameters and protection functions.

The determination of pressure, temperature, power and RCS flow uncertainties are applicable for the Point Beach Nuclear Plant Units 1 & 2 for power levels up to 1656 Mwt - NSSS power, for 18 month fuel cycles + 25% per the plant Technical Specifications, and for a full power Tavg window of 558.1 to 574.0°F. These uncertainties are also applicable for power levels up to 1679 Mwt - NSSS power, and for a full power Tavg window of 558.6 to 573.4°F, when the daily calorimetric power measurement is based on the LEFM on the feedwater header.

II. METHODOLOGY

The methodology used to combine the error components for a channel is the square root of the sum of the squares of those groups of components which are statistically independent. Those errors that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two sided distributions. The sum of both sides is equal to the range for that parameter, e.g., Rack Drift is typically [ ]+a,c, the range for this parameter is [ ]+a,c. This technique has been utilized before as noted above, and has been endorsed by the NRC staff[7,8,9,10] and various industry standards[11,12].

The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse Setpoint Methodology[13] and are defined as follows:

1. For precision parameter indication using Special Test Equipment or a digital voltmeter (DVM) at the input to the racks;

$$CSA = \{(SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2 + (RDOUT)^2\}^{1/2} + BIAS \qquad \text{Eq. 1}$$

2. For parameter indication utilizing the plant process computer;

$$CSA = \{(SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2 + (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + (RMTE + A/D)^2\}^{1/2} + BIAS \qquad \text{Eq. 2}$$

3. For parameters with closed-loop automatic control systems, the calculation takes credit for [ ]+a,c. There is a functional dependency between the transmitters/racks and the automatic control system/indicator when an uncertainty in the transmitters/racks is common to the automatic control system and the indicator. That is, an uncertainty in the high direction in the transmitter/ racks will result in a high uncertainty in the automatic control system and the indicator. To account for the functional dependency, a square root function is used for the transmitter/ racks/reference signal, and a square root function is used for the controller/indicators;

$$CSA = \{(PMA^2(random) + (PEA)^2$$

$$+ (SMTE + SCA)^2 + (SPE)^2 + (STE)^2 + (SMTE + SD)^2 + (SRA)^2$$

$$+ (RMTE + RCA)^2 + (RTE)^2 + (RMTE + RD)^2 + (REF)^2\}^{1/2}$$

$$+ \{(CMTE + CA)^2 + (RMTE + RCA)^2_{IND} + (RDOUT)^2_{IND}\}^{1/2}$$

$$+ BIAS \qquad \text{Eq. 3}$$

where:

CSA	=	Channel Statistical Allowance
PMA	=	Process Measurement Accuracy
PEA	=	Primary Element Accuracy
SRA	=	Sensor Reference Accuracy
SCA	=	Sensor Calibration Accuracy
SMTE	=	Sensor Measurement and Test Equipment Accuracy
SPE	=	Sensor Pressure Effects
STE	=	Sensor Temperature Effects
SD	=	Sensor Drift
RCA	=	Rack Calibration Accuracy
RMTE	=	Rack Measurement and Test Equipment Accuracy
RTE	=	Rack Temperature Effects
RD	=	Rack Drift
RDOUT	=	Readout Device Accuracy
CA	=	Controller Allowance
CMTE	=	Controller Measurement and Test Equipment Acuracy
A/D	=	Analog to Digital Conversion Accuracy
REF	=	Reference signal for automatic control sytem
IND	=	Indicator.

PMA and PEA terms are not included in equations 1 and 2 since the equations are to determine instrumentation uncertainties only. PMA and PEA terms are included in the determination of control system uncertainties.

The parameters above are defined in references 5 and 12 and are based on SAMA Standard PMC 20.1, 1973[14]. However, for ease in understanding they are paraphrased below:

PMA - non-instrument related measurement errors, e.g., temperature stratification of a fluid in a pipe.
PEA - errors due to a metering device, e.g., elbow, venturi, orifice.
SRA - reference (calibration) accuracy for a sensor/transmitter.
SCA - calibration tolerance for a sensor/transmitter.
SMTE - measurement and test equipment used to calibrate a sensor/transmitter.
SPE - change in input-output relationship due to a change in static pressure

for a differential pressure (d/p) transmitter.

STE - change in input-output relationship due to a change in ambient temperature for a sensor or transmitter.

SD - change in input-output relationship over a period of time at reference conditions for a sensor or transmitter.

RCA - calibration accuracy for all rack modules in loop or channel assuming the loop or channel is string calibrated, or tuned, to this accuracy.

RMTE - measurement and test equipment used to calibrate rack modules.

RTE - change in input-output relationship due to a change in ambient temperature for the rack modules.

RD - change in input-output relationship over a period of time at reference conditions for the rack modules.

RDOUT - the measurement accuracy of a special test local gauge, digital voltmeter or multimeter on it's most accurate applicable range for the parameter measured, or 1/2 the smallest increment on an indicator (readability).

CA - allowance of the controller rack module(s) that performs the comparison and calculates the difference between the controlled parameter and the reference signal.

CMTE - measurement and test equipment used to calibrate the controller rack module(s) that perform(s) the comparison between the controlled parameter and the reference signal.

A/D - allowance for conversion accuracy of an analog signal to a digital signal for process computer use.

REF - the reference signal uncertainty for a closed-loop automatic control system.

IND - indicator accuracies are used for these uncertainty calculations. Control board indicators are typically used.

BIAS - a one directional uncertainty for a sensor/transmitter or a process parameter with a known magnitude.

A more detailed explanation of the Westinghouse methodology noting the interaction of several parameters is provided in references 6 and 13.

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III. INSTRUMENTATION UNCERTAINTIES

The instrumentation uncertainties will be discussed first for the two parameters which are controlled by automatic systems, Pressurizer Pressure, and T_{avg} (through automatic rod control).

1. PRESSURIZER PRESSURE

Pressurizer pressure is controlled by a closed-loop automatic control system that compares the measured vapor space pressure to a reference value. This uncertainty calculation takes credit for the closed-loop control system design where [$]^{+a,c}$. The control channel uncertainties for the automatic control system include allowances for the pressure transmitters, the process racks, and the control system reference setpoint. The pressurizer pressure control system reference setpoint is generated by the setting of a variable potentiometer on the Main Control Board manual/automatic station. The reference setpoint (Pref) is adjusted and verified by the plant operators with the control board indicators. This uncertainty calculation also includes the control board indicators for verification of the automatic control system performance.

On Table 1, the calculated electronics uncertainty for this function using Equation 3 is [$]^{+a,c}$ with a [$]^{+a,c}$ bias corresponding to [$]^{+a,c}$ with a [$]^{+a,c}$ bias for the average of 3 control board indicators. In addition to the control system uncertainty, an allowance is made for pressure overshoot or undershoot due to the interaction and thermal inertia of the heaters and spray. An allowance of [$]^{+a,c}$ is made for this effect. The total control system uncertainty including indication is [$]^{+a,c}$ with a [$]^{+a,c}$ bias which results in a standard deviation of [$]^{+a,c}$ for a normal, two sided probability distribution.

TABLE 1
PRESSURIZER PRESSURE CONTROL SYSTEM UNCERTAINTY

All Values in % Span

+a,c

REF	=	
PMA	=	
PEA	=	
SRA	=	
SCA	=	
SMTE	=	
STE	=	
SD	=	
BIAS	=	
RCA	=	
RMTE	=	
RTE	=	
RD	=	
RCA_{IND}	=	
$RMTE_{IND}$	=	
$RDOUT_{IND}$	=	
CA	=	
CMTE	=	

RANGE = 1700 - 2500 psig, SPAN = 800 psi
CHANNELS P-429, -430, -431 & -449

+a,c

ELECTRONICS UNCERTAINTY =
PLUS

ELECTRONICS UNCERTAINTY =
PLUS

CONTROLLER UNCERTAINTY =

* 15 psi setting tolerance around 2235 psig

2. <u>Tavg</u>

T_{avg} is controlled by a system that compares the high T_{avg} from the loops with a reference derived from the First Stage Turbine Impulse Chamber Pressure. T_{avg} is the average of the narrow range T_H and T_C values. The high loop T_{avg} is then used for rod control. Allowances are made (as noted on Table 2) for the RTDs, transmitter and the process racks/indicators and controller. The CSA for this function is dependent on the type of RTD, pressure transmitter, and the location of the RTDs, i.e., in the Hot and Cold Leg bypass manifolds. Based on one T_H and one T_C RTD per channel to calculate T_{avg} and with the RTDs located in the hot and cold leg bypass manifolds, the calculated CSA for the electronics using Equation 3 is [ ]$^{+a,c}$. Assuming a normal, two sided probability distribution results in an electronics standard deviation (s_1) of [ ]$^{+a,c}$.

However, this does not include the deadband of ±1.5°F for automatic control. The T_{avg} controller accuracy is the combination of the instrumentation accuracy and the deadband. The probability distribution for the deadband has been determined to be [ ].$^{+a,c}$ The variance for the deadband uncertainty is then:

$(s_2)^2$ = [ ]$^{+a,c}$ = [ ]$^{+a,c}$

where [ ]$^{+a,c}$. Combining the variance for instrumentation and deadband results in a controller variance of:

$(s_T)^2 = (s_1)^2 + (s_2)^2$ = [ ]$^{+a,c}$

The controller s_T = [ ]$^{+a,c}$ for a total random uncertainty of [ ]$^{+a,c}$.

An additional bias of [ ]$^{+a,c}$ for T_{cold} streaming (in terms of Tavg) based on a conservative [ ]$^{+a,c}$ T_{cold} streaming uncertainty is included in Table 2. An additional bias of [ ]$^{+a,c}$ for R/E (resistance to voltage [or electromagnetic force]) linearization (in terms of Tavg) is included in Table 2. Therefore, the total uncertainty of the controller with the additional biases is [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ bias.

TABLE 2
TAVG ROD CONTROL SYSTEM UNCERTAINTY

		% T_{avg} Span	% Turbine Pressure Span (P-485,-486)
PMA_1	=		
PMA_2	=		
SRA	=		
SCA	=		
SMTE	=		
STE	=		
SD	=		
$BIAS_1$	=		
$BIAS_2$	=		
R/E	=		
R/E_MTE	=		
RCA	=		
RMTE	=		
RTE	=		
RD	=		
RCA_{IND}	=		
$RMTE_{IND}$	=		
$RDOUT_{IND}$	=		
CA	=		
CMTE	=		

+a,c

Hot Leg RTDs = 1/Channel #Cold Leg RTDs = 1/Channel

Tavg span = 100 °F (520-620°F)

R/E span = 150 °F (500-650°F for Hot and Cold Leg)

Turbine pressure span = 650psi (0-650 psig)

ELECTRONICS CSA	=	
ELECTRONICS SIGMA	=	
CONTROLLER SIGMA	=	
CONTROLLER CSA	=	
CONTROLLER BIAS	=	

+a,c

Note A: Module TM-401BB = 0.5% span
Modules for Loop A1 similar for A2, B1 & B2.

Note B: Module PM-485A = 0.5% span

Note C: Module TM-401EE = 0.5% span
Module TM-401D = 0.5% span
Module TM-401H = 0.5% span
Module TM-401M = 0.25% span
Module TM-401P = 0.82% span
Module TM-401N = 1.63% span
Module TM-401I = 0.5% span

3. RCS FLOW

3.1 Calorimetric RCS Flow Measurement Uncertainty (Using LEFM on the Feedwater Header)

RTDP and Point Beach's Technical Specifications require an RCS flow measurement with a high degree of accuracy. A total RCS flow measurement every fuel cycle, typically 18 months, is performed to verify RCS flow and to normalize the RCS flow instrument channels. Interim surveillances performed with the process computer ensure that the RCS flow is maintained within the assumed safety analysis values, i.e., Minimum Measured Flow (MMF). The 18 month RCS flow surveillance is satisfied by a secondary side power-based calorimetric RCS flow measurement. The calorimetric flow measurement is performed at the beginning of a cycle near full power operation.

Eighteen month instrument drift is used in this uncertainty analysis for hot and cold leg RTDs, and for feedwater pressure, steam pressure and pressurizer pressure transmitters.

A Leading Edge Flow Meter (LEFM) installed on the Feedwater header is used to determine total Feedwater flow. Feedwater temperature indication by the LEFM is compared to individual loop feedwater temperatures which are then adjusted if necessary.

The flow measurement is performed by determining the Steam Generator thermal output (corrected for the RCP heat input and the loop's share of primary system heat losses) and the enthalpy rise (Delta-h) of the primary coolant. Assuming that the primary and secondary sides are in equilibrium, the RCS total vessel flow is the sum of the individual primary loop flows, i.e.,

$$W_{RCS} = \Sigma_{i=1}{}^{N}(W_L)_i. \qquad \text{Eq. 4}$$

The individual primary loop volumetric flows are determined by correcting the thermal output of the Steam Generator for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the loop's share of the primary side system losses, dividing by the primary side enthalpy rise and multiplying by the Cold Leg specific volume. The equation for this calculation is:

$$W_L = \frac{(A)\{Q_{SG} - Q_P + (Q_L/N)\}(V_C)}{(h_H - h_C)} \qquad \text{Eq. 5}$$

where;

W_L	=	Loop Flow (gpm)
A	=	Constant conversion factor 0.1247 gpm/(ft^3/hr)
Q_{SG}	=	Steam Generator thermal output (BTU/hr)
Q_P	=	RCP heat addition (BTU/hr)
Q_L	=	Primary system net heat losses (BTU/hr)

V_C = Specific volume of the Cold Leg at T_C (ft^3/lb)
N = Number of primary side loops
h_H = Hot Leg enthalpy (BTU/lb)
h_C = Cold Leg enthalpy (BTU/lb).

The thermal output of the Steam Generator is determined by a secondary side calorimetric measurement, which is defined as:

$$Q_{SG} = (h_s - h_f)W_f \qquad \text{Eq. 6}$$

where;
h_s = Steam enthalpy (BTU/lb)
h_f = Feedwater enthalpy (BTU/lb)
W_f = Feedwater flow (LEFM feedwater header flow divided by # loops)(lb/hr).

The steam enthalpy is based on the measurement of steam generator outlet steam pressure assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and nominal feedwater pressure. The feedwater flow is determined by LEFM measurements.

RCP heat addition is determined by calculation, based on the best estimate of coolant flow, pump head, and pump hydraulic efficiency.

The primary system net heat losses are determined by calculation, considering the following system heat inputs (+) and heat losses (-):

Charging flow (+)
Letdown flow (-)
Seal injection flow (+)
RCP thermal barrier cooler heat removal (-)
Pressurizer spray flow (-)
Pressurizer surge line flow (+)
Component insulation heat losses (-)
Component support heat losses (-)
CRDM heat losses (-).

A single calculated sum for 100% Rated Thermal Power (RTP) operation is used for these losses or heat inputs.

The Hot Leg and Cold Leg enthalpies are based on the measurement of the Hot Leg temperature, Cold Leg temperature and the nominal Pressurizer pressure. The Cold Leg specific volume is based on measurement of the Cold Leg temperature and nominal Pressurizer pressure.

The RCS flow measurement is thus based on the following plant measurements:

- Steamline pressure (P_S)
- Feedwater temperature (T_f)
- Feedwater pressure (P_f)
- Feedwater flow from LEFM
- Hot Leg temperature (T_H)
- Cold Leg temperature (T_C)
- Pressurizer pressure (P_p)
- Steam Generator blowdown flow (if not secured)

and on the following calculated values:

- Feedwater density (ρ_f)
- Feedwater enthalpy (h_f)
- Steam enthalpy (h_S)
- Moisture carryover (impacts h_S)
- Primary system net heat losses (Q_L)
- RCP heat addition (Q_p)
- Hot Leg enthalpy (h_H)
- Cold Leg enthalpy (h_C).

These measurements and calculations are presented schematically in Figure 1. The derivation of the measurement and flow uncertainties on Table 5 are noted below.

<u>Secondary Side</u>

The secondary side uncertainties are in four principal areas, Feedwater flow, Feedwater enthalpy, Steam enthalpy and net pump heat addition. These areas are specifically identified on Table 5.

For the measurement of Feedwater flow, the LEFM is located on the feedwater header and provides a total flow. The accuracy to which the total flow is determined is based on calculations performed by the manufacture of the LEFM.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in Feedwater temperature and pressure. Table 3 notes the instrument uncertainties for the hardware used to perform the measurements. Table 4 lists the various sensitivities. As can be seen on Table 5, Feedwater temperature uncertainties have an impact on

Feedwater density and Feedwater enthalpy. Feedwater pressure uncertainties impact Feedwater density and Feedwater enthalpy.

Using the NBS/NRC Steam Tables, it is possible to determine the sensitivity of Steam enthalpy to changes in Steam pressure and Steam quality. Table 3 notes the uncertainty in Steam pressure and Table 4 provides the sensitivity. For Steam quality, the Steam Tables were used to determine the sensitivity at a moisture content of [ ]$^{+a,c}$. This value is noted on Table 4.

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and are summarized for a two loop plant as follows:

System heat losses	- 2.0 MWt
Component conduction and convection losses	- 1.4 MWt
Pump heat adder	+ 9.4 MWt
Net Heat input to RCS	+ 6.0 MWt

The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]$^{+a,c}$ of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]$^{+a,c}$ of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2 and by input power measurements from several other plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is [ ]$^{+a,c}$ of core power.

<u>Primary Side</u>

The primary side uncertainties are in three principal areas, hot leg enthalpy, cold leg enthalpy and cold leg specific volume. These are specifically noted on Table 5. Three primary side parameters are actually measured, T_H, T_C and Pressurizer pressure. Hot Leg enthalpy is influenced by T_H, Pressurizer pressure and Hot Leg temperature streaming. The uncertainties for the instrumentation are noted on Table 3 and the sensitivities are provided on Table 4. The hot leg streaming is split into random and systematic components. For Point Beach Units 1 & 2 where the RTDs are located in bypass manifolds, the hot leg temperature streaming uncertainty components are [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ systematic.

The cold leg enthalpy and specific volume uncertainties are impacted by T_C and Pressurizer pressure. Table 3 notes the T_C instrument uncertainty and Table 4 provides the sensitivities.

Parameter dependent effects are identified on Table 5. Westinghouse has determined the dependent sets in the calculation and the direction of interaction, i.e., whether components in a dependent set are additive or subtractive with respect to a conservative calculation of RCS flow. The same work was performed for the instrument bias values. As a result, the calculation explicitly accounts for dependent effects and biases with credit taken for sign (or direction of impact).

Using Table 5, the 2 loop uncertainty equation (with biases) is as follows:

+a,c

Based on the number of loops; number, type and measurement method of RTDs, and the vessel Delta-T, the flow uncertainty is:

# of loops	flow uncertainty (% flow)
2	[ ]+a,c

TABLE 3

FLOW CALORIMETRIC INSTRUMENTATION UNCERTAINTIES
(% SPAN)

	FW TEMP	FW PRES	FW FLOW	STM PRESS	TH	TC	PRZ PRESS	
								+a,c
LEFM =								
SRA =								
SCA =								
SMTE =								
SPE =								
STE =								
SD =								
BIAS =								
R/E =								
RMTE =								
RTE =								
RD =								
RDOUT =								
CSA =								
# OF INSTRUMENTS USED	1	1/Loop	1	1/Loop	2/Loop	2/Loop	4	
	°F	psi	% Flow	psi	°F	°F	psi	
INST SPAN =	(1)	1500(2)	100(3)	1400(4)	150(5)	150(5)	800(6)	
								+a,c
INST UNC. (RANDOM) =								
INST UNC. (BIAS) =								
NOMINAL =	442.9 °F	737-864 psia	100 % Flow	637-764 psia	591.2-605.5 °F	526.0-541.4 °F	2250 psia	

(1) Special test instrumentation TE-3111 and an Resistance Thermometer bridge are used for this measurement.
(2) Pressure (P-2245) is measured with a digital voltmeter at the input of the process instrumentation.
(3) Flow (F-3110) is measured with an LEFM on the feedwater header.
(4) Pressure (P-468, -469, -478, -479, -482, -483) is measured with a digital voltmeter at the input of the process instrumentation.
(5) Temperature is measured with a digital voltmeter at the output of the R/E process instrumentation modules.
(6) Pressure (P-429, -430, -431, -449) is measured with a digital voltmeter at the input of the process instrumentation.
(7) Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246).

TABLE 4

FLOW CALORIMETRIC SENSITIVITIES

+a,c

FEEDWATER FLOW	
LEFM	=
DENSITY	
TEMPERATURE	=
PRESSURE	=
FEEDWATER ENTHALPY	
TEMPERATURE	=
PRESSURE	=
h_s	=
h_f	=
Dh (SG)	=
STEAM ENTHALPY	
PRESSURE	=
MOISTURE	=
HOT LEG ENTHALPY	
TEMPERATURE	=
PRESSURE	=
h_H	=
h_C	=
Dh (VESS)	=
COLD LEG ENTHALPY	
TEMPERATURE	=
PRESSURE	=
COLD LEG SPECIFIC VOLUME	
TEMPERATURE	=
PRESSURE	=

* Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246).

TABLE 5

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTY

+a,c

COMPONENT	INSTRUMENT UNCERTAINTY	FLOW UNCERTAINTY
FEEDWATER FLOW		
LEFM		
DENSITY		
TEMPERATURE		
PRESSURE		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
HOT LEG ENTHALPY		
TEMPERATURE		
STREAMING, RANDOM		
STREAMING, SYSTEMATIC		
PRESSURE		
COLD LEG ENTHALPY		
TEMPERATURE		
PRESSURE		
COLD LEG SPECIFIC VOLUME		
TEMPERATURE		
PRESSURE		

*, **, +, ++ Indicates Sets of Dependent Paremeters
*** Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246)

TABLE 5 (CONTINUED)

CALORIMETRIC RCS FLOW MEASUREMENT UNCERTAINTY

COMPONENT		FLOW UNCERTAINTY
		+a,c
BIAS VALUES		
COLD LEG ENTHALPY TEMPERATURE		
COLD LEG SPECIFIC VOLUME TEMPERATURE		
FLOW BIAS TOTAL VALUE		
		+a,c
2 LOOP UNCERTAINTY	(WITHOUT BIAS VALUES)	
2 LOOP UNCERTAINTY	(WITH BIAS VALUES)	

3.2 Loop RCS Flow Uncertainty (Using Plant Computer)

As noted earlier, the calorimetric RCS flow measurement is used as the reference for normalizing the loop RCS flow measurement from the cold leg elbow tap transmitters. Since the cold leg elbow tap transmitters feed the plant computer, it is a simple matter to perform an RCS flow surveillance. Table 6 notes the instrument uncertainties for determining flow by using the loop RCS flow channels and the plant computer, assuming one loop RCS flow channel per reactor coolant loop. The d/p transmitter uncertainties are converted to percent flow using the following conversion factor:

$$\% \text{ flow} = (\text{d/p uncertainty})(1/2)(\text{FLOWmax} / \text{FLOWnominal})^2$$

where FLOWmax is the maximum value of the loop RCS flow channel. The loop RCS flow uncertainty is then combined with the calorimetric RCS flow measurement uncertainty. This combination of uncertainties results in the following total flow uncertainty:

# of loops	flow uncertainty (% flow)
2	± 2.06 random + 0.25 bias

The corresponding value used in RTDP is:

# of loops	standard deviation (% flow)
2	[ ] +a,c

TABLE 6

LOOP RCS FLOW UNCERTAINTY
PLANT COMPUTER

INSTRUMENT UNCERTAINTIES
1 LOOP RCS FLOW CHANNEL PER REACTOR COOLANT LOOP
(F-411, -412, -413, -414, -415, -416)

	% d/p SPAN	% Flow
PMA =		
PEA =		
SRA =		
SCA =		
SMTE =		
SPE =		
STE =		
SD =		
BIAS =		
RCA =		
RMTE =		
RTE =		
RD =		
A/D =		
A/D_MTE=		
FLOW CALORIMETRIC BIAS =		
FLOW CALORIMETRIC =		
INSTRUMENT SPAN =		

+a,c

SINGLE LOOP ELBOW TAP FLOW UNCERTAINTY = [ ] +a,c

2 LOOP RCS FLOW UNCERTAINTY (WITHOUT BIAS VALUES) = 2.06% flow

2 LOOP RCS FLOW UNCERTAINTY (WITH BIAS VALUES) = 2.3% flow

Note A: Module FM-411, -412, -413, -414, -415, -416 = 0.5% span
* Zero values due to normalization to calorimetric RCS flow measurement

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4. REACTOR POWER (using feedwater venturis)

The plant is required to perform a primary/secondary side heat balance at least every 24 hours when power is above 15% Rated Thermal Power. This heat balance is used to verify that the plant is operating within the limits of the Operating License (1650 Mwt-Core power) or 1656 Mwt-NSSS power when using the feedwater venturis, and to adjust the Power Range Neutron Flux channels when the difference between the Power Range Neutron Flux channels and the heat balance is greater than allowed by the plant Technical Specifications. PBNP also continuously calculates the reactor thermal output (RTO) to ensure that the power limit is not exceeded.

Assuming that the primary and secondary sides are in equilibrium; the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core Btu/hr at rated full power. The equation for this calculation is:

$$RP = \frac{\{\sum_{i=1}^{N}\{Q_{SG} - Q_P + (Q_L/N)\}_i\}(100)}{H} \qquad \text{Eq. 7}$$

where;

RP = Core power (% RTP)
N = Number of primary side loops
Q_{SG} = Steam generator thermal output (BTU / hr) as defined in Eq. 6
Q_P = RCP heat addition (BTU / hr) as defined in Eq. 5
Q_L = Primary system net heat losses (BTU / hr) as defined in Eq. 5
H = Rated core power (BTU / hr).

For the purposes of this uncertainty analysis (and based on H noted above) it is assumed that the plant is at 100% RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values. However, operation at lower power levels results in increased margin to DNB far in excess of any margin losses due to increased measurement uncertainty.

The feedwater flow in equation 6 is determined by multiple measurements and the following calculation:

$$W_f = (K)(F_a)\{(\rho_f)(d/p)\}^{1/2} \qquad \text{Eq. 8}$$

where:

W_f = Feedwater loop flow (lb/hr)
K = Feedwater venturi flow coefficient
F_a = Feedwater venturi correction for thermal expansion
ρ_f = Feedwater density (lb/ft^3)
d/p = Feedwater venturi pressure drop (inches H_2O).

The feedwater venturi flow coefficient is the product of a number of constants including as-built dimensions of the venturi and calibration tests performed by the vendor. The thermal expansion correction is based on the coefficient of expansion of the venturi material and the difference between feedwater temperature and calibration temperature. Feedwater density is based on the measurement of feedwater temperature and feedwater pressure. The venturi pressure drop is obtained from the output of the differential pressure transmitter connected to the venturi.

The power measurement is thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f)
Feedwater pressure (P_f)
Feedwater venturi differential pressure (d/p)
Steam generator blowdown (if not secured);

and on the following calculated values:

Feedwater venturi flow coefficients (K)
Feedwater venturi thermal expansion correction (F_a)
Feedwater density (ρ_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Moisture carryover (impacts h_s)
Primary system net heat losses (Q_L)
RCP heat addition (Q_p)

Secondary Side

The secondary side power calorimetric equations and effects are the same as those noted for the calorimetric RCS flow measurement (secondary side portion), equation 6. The measurements and calculations are presented schematically on Figure 2.

For the measurement of feedwater flow, each feedwater venturi is calibrated by the vendor in a hydraulics laboratory under controlled conditions to an accuracy of [ ]$^{+a,c}$. The calibration data that substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [ ]$^{+a,c}$ is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [ ]$^{+a,c}$. Since the calculated steam generator thermal output is proportional to feedwater flow, the flow coefficient uncertainty is expressed as [ ]$^{+a,c}$. It should be noted that no allowance is made for feedwater venturi fouling. The effect of fouling results in an indicated power higher than actual, which is conservative.

The uncertainty applied to the feedwater venturi thermal expansion correction (F_a) is based on the uncertainties of the measured feedwater temperature and the coefficient of thermal expansion for the venturi material, 304 stainless steel. For this material, a change of ± 1.0 °F in the nominal feedwater temperature range changes F_a by [ ]$^{+a,c}$ and the steam generator thermal output by the same amount.

Based on data introduced into the ASME Code, the uncertainty in F_a for 304 stainless steel is ± 5 %. This results in an additional uncertainty of [ ]$^{+a,c}$ in power.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 7 notes the instrument uncertainties for the hardware used to perform the measurements. Table 8 lists the various sensitivities. As can be seen on Table 8, feedwater temperature uncertainties have an impact on venturi F_a, feedwater density and feedwater enthalpy. Feedwater pressure uncertainties impact feedwater density and feedwater enthalpy.

Feedwater venturi d/p uncertainties are converted to % feedwater flow using the following conversion factor:

% flow = (d/p uncertainty)(1/2)(FLOWmax/FLOWnominal)^2.

The feedwater flow transmitter span (FLOWmax) is 120.0% of nominal flow.

Since it is necessary to make this determination daily, the plant computer is used for the calorimetric power measurement. As noted in Table 9, Westinghouse has determined the dependent sets in the calculation and the direction of interaction. This is the same as that performed for the calorimetric RCS flow measurement, but applicable only to power. The same was performed for the bias values.

Using the power uncertainty values noted on Table 9, the 2 loop uncertainty (with bias values) equation is as follows:

+a,c

[ ]

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement using the feedwater flow venturis is:

# of loops	power uncertainty (% power)
2	[ ] +a,c

TABLE 7

POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES

(% SPAN)

	FW TEMP	FW PRES	FW D/P	STM PRESS
				+a,c
SRA =				
SCA =				
SMTE =				
SPE =				
STE =				
SD =				
BIAS =				
RCA =				
RMTE =				
RTE =				
RD =				
A/D =				
CSA =				

	FW TEMP	FW PRES	FW D/P	STM PRESS
# OF INSTRUMENTS USED	1/Loop	1/Loop	1/Loop	1/Loop
	°F	psi	% d/p	psi
INST SPAN =	150	1600	120% Flow	1400
				+a,c
INST UNC. (RANDOM) =				
INST UNC. (BIAS) =				
NOMINAL =	440.7 °F	800-875 psia	100 % Flow	700-775 psia

(a) Included in RCA

Feedwater temperature measurement is from channels T-2104 and -2105

Feedwater pressure measurement is from channels P-2289 and -2290

Feedwater flow measurement is from channels F-466, -467, -476 and -477

Steam pressure measurement is from channels P-468, -469, -478, -479, -482 and -483.

TABLE 8

POWER CALORIMETRIC SENSITIVITIES

+a,c

FEEDWATER FLOW		
F_a		
TEMPERATURE	=	
MATERIAL	=	
DENSITY		
TEMPERATURE	=	
PRESSURE	=	
DELTA P	=	
FEEDWATER ENTHALPY		
TEMPERATURE	=	
PRESSURE	=	
h_s	=	
h_f	=	
Dh (SG)	=	
STEAM ENTHALPY		
PRESSURE	=	
MOISTURE	=	

TABLE 9

SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

COMPONENT	INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY
		+a,c
FEEDWATER FLOW		
VENTURI		
THERMAL EXPANSION COEFFICIENT		
TEMPERATURE		
MATERIAL		
DENSITY		
TEMPERATURE		
PRESSURE		
DELTA P		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
BIAS VALUES		
FEEDWATER DELTA P		
POWER BIAS TOTAL VALUE		
SINGLE LOOP UNCERTAINTY (WITHOUT BIAS)		
2 LOOP UNCERTAINTY (WITHOUT BIAS)		
2 LOOP UNCERTAINTY (WITH BIAS VALUES)		

*, * *, INDICATES SETS OF DEPENDENT PARAMETERS

5. <u>REACTOR POWER (using LEFM on feedwater header)</u>

The plant performs a primary/secondary side heat balance at least every 24 hours when power is above 15% Rated Thermal Power. This heat balance is used to verify that the plant is operating within the limits of the Operating License (1673 Mwt-Core power) or 1679 Mwt-NSSS power when using the LEFM on the feedwater header, and to adjust the Power Range Neutron Flux channels when the difference between the Power Range Neutron Flux channels and the heat balance is greater than allowed by the plant Technical Specifications. PBNP also continuously calculates the reactor thermal output (RTO) to ensure that the power limit is not exceeded.

Assuming that the primary and secondary sides are in equilibrium; the core power is determined by summing the thermal output of the steam generators, correcting the total secondary power for Steam Generator blowdown (if not secured), subtracting the RCP heat addition, adding the primary side system losses, and dividing by the core Btu/hr at rated full power. Equation 7 is used for this calculation.

For the purposes of this uncertainty analysis it is assumed that the plant is at 100% RTP when the measurement is taken. Measurements performed at lower power levels will result in different uncertainty values. However, operation at lower power levels results in increased margin to DNB far in excess of any margin losses due to increased measurement uncertainty.

The thermal output of the Steam Generator is determined by a secondary side calorimetric measurement which is defined by Equation 9 as:

$$Q_{SG} = (h_s - h_f)W_f - (h_s - h_{sgbd})W_{sgbd} \qquad \text{Eq. 9}$$

where;

h_s = Steam enthalpy (BTU/lb)

h_f = Feedwater enthalpy (BTU/lb)

h_{sgbd} = Steam generator blowdown enthalpy (BTU/lb)

W_f = Feedwater flow (LEFM feedwater header flow divided by # loops)(lb/hr)

W_{sgbd} = Steam generator blowdown flow (lb/hr).

The steam enthalpy is based on the measurement of steam generator outlet steam pressure, assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and feedwater pressure. The feedwater flow and feedwater temperature are determined by a Leading Edge Flow Meter (LEFM) measurement on the main feedwater header, and it is assumed that the loop feedwater flows are equal.

The steam generator blowdown flow is the outlet flow from the steam generators used to control water chemistry, and is determined by measurement from the steam generator loop blowdown flow orifice and the following calculation:

$$W_{sgbd} = (K)(F_a)(a)\ \{(2)(g_c)(p_f)(d/p)\}^{1/2} \qquad \text{Eq.10}$$

where;

	K	=	Steam generator loop blowdown flow orifice coefficient
	F_a	=	Steam generator loop blowdown flow orifice correction for thermal expansion
	a	=	Steam generator loop blowdown flow orifice area
	g_c	=	Gravitational constant (32.174 ft/sec^2)
	p_f	=	Steam generator loop blowdown flow density (lb/ft^3)
	d/p	=	Steam generator loop blowdown flow orifice pressure drop (inches H_2O).

The steam generator blowdown orifice flow coefficient is the product of a number of constants including as-built dimensions of the orifice and pipe internal diameter. The thermal expansion correction is based on the coefficient of expansion of the orifice material and the difference between steam generator blowdown temperature and calibration temperature. Steam generator blowdown density and enthalpy are based on the measurement of steam generator steam pressure. The blowdown liquid enthalpy is assumed to be equal to that of a saturated liquid at the measured steam pressure. The orifice pressure drop is obtained from the output of the differential pressure indicator.

RCP heat addition is determined by calculation, based on the best estimate of coolant flow, pump head, and pump hydraulic efficiency.

The primary system net heat losses are determined by calculation, considering the previously defined system heat inputs (+) and heat losses (-).

A single calculated sum for 100% RTP operation is used for these losses or heat inputs.

The power measurement is thus based on the following plant measurements:

Steamline pressure (P_s)
Feedwater temperature (T_f - from LEFM)
Feedwater pressure (P_f)
Feedwater header flow (from LEFM)
Steam generator loop blowdown flow orifice differential pressure (d/p)(if not secured);

and on the following calculated values:

Steam generator loop blowdown flow orifice coefficient (K)
Steam generator loop blowdown flow orifice thermal expansion correction (F_a)
Steam generator loop blowdown flow orifice area (a)
Feedwater density (p_f)
Feedwater enthalpy (h_f)
Steam enthalpy (h_s)
Steam generator blowdown enthalpy (h_{sgbd})
Steam generator blowdown density (p_f)

Moisture carryover (impacts h_s)
Primary system net heat losses (Q_L)
RCP heat addition (Q_p)

The derivation of the measurement uncertainties and the calorimetric power measurement uncertainties on Table 12 are noted below.

Secondary Side

The secondary side power calorimetric equations and effects are the same as those noted for the calorimetric RCS flow measurement (secondary side portion), equation 6. The measurements and calculations are presented schematically on Figure 3.

For the measurement of feedwater flow and feedwater temperature, an LEFM is installed on the feedwater header, and the accuracies have been provided by Wisconsin Electric and Caldon, Inc.

Using the NBS/NRC Steam Tables it is possible to determine the sensitivities of various parameters to changes in feedwater temperature and pressure. Table 10 notes the instrument uncertainties for the hardware used to perform the measurements. Table 11 lists the various sensitivities. As can be seen on Table 12, feedwater temperature uncertainties have an effect on feedwater density and feedwater enthalpy. Feedwater pressure uncertainties affect feedwater density and feedwater enthalpy.

Using the NBS/NRC Steam Tables again, it is possible to determine the sensitivity of steam enthalpy to changes in steam pressure and steam quality. Table 10 notes the uncertainty in steam pressure and Table 11 provides the sensitivity. For steam quality, the NBS/NRC Steam Tables were used to determine the sensitivity for a moisture content of [ ]$^{+a,c}$, and the value associated with the limiting power measurement uncertainty is noted on Table 11.

The net pump heat addition uncertainty is derived from the combination of the primary system net heat losses and pump heat addition and was previously summarized as noted for the calorimetric RCS flow measurement (secondary side portion):

Since it is necessary to make this determination daily, the plant computer is used for the calorimetric power measurement. As noted in Table 12, Westinghouse has determined the dependent sets in the calculation and the direction of interaction. This is the same as that performed for the calorimetric RCS flow measurement, but applicable only to power. The same was performed for the bias values.

Using the power uncertainty values noted on Table 12, the 2 loop uncertainty (with bias values) equation is as follows:

+a,c

[ ]

Based on the number of loops and the instrument uncertainties for the four parameters, the uncertainty for the secondary side power calorimetric measurement is:

# of loops	power uncertainty (% power)
2	[ ] +a,c

TABLE 10
POWER CALORIMETRIC INSTRUMENTATION UNCERTAINTIES
(% SPAN)

+a,c

	FW TEMP (HEADER)	FW PRES	FW FLOW (HEADER)	STM PRESS	SG BLOWDOWN FLOW
LEFM =					
SRA =					
SCA =					
SMTE =					
SPE =					
STE =					
SD =					
BIAS =					
RCA =					
RMTE =					
RTE =					
RD =					
A/D =					
RDOUT=					
CSA =					
# OF INSTRUMENTS USED	1/Header	1/Loop	1/Header	1/Loop	1/Loop
	°F	psi	% flow	psi	% flow
INST SPAN =		1600		1400	1.35% rated feedwater flow(rfwf)
INST UNC. (RANDOM) =					
INST UNC. (BIAS) =					
NOMINAL =	442.9 °F	737-864 psia	100 % Flow	637-764 psia	15000 lb/hr/loop

+a,c

(a) Included in RCA
Feedwater temperature measurement is from the Caldon LEFM
Feedwater pressure measurement is from channels P-2289 and -2290
Feedwater flow measurement is from the Caldon LEFM
Steam pressure measurement is from channels P-468, -469, -478, -479, -482 and -483.
Steam generator loop blowdown flow is from channels F-5940 and-5941 (range: 0-50000 lb/hr)
* Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246)

TABLE 11

POWER CALORIMETRIC SENSITIVITIES

+a,c

FEEDWATER DENSITY	
TEMPERATURE	=
PRESSURE	=
FEEDWATER ENTHALPY	
TEMPERATURE	=
PRESSURE	=
h_s	=
h_f	=
Dh (SG)	=
STEAM ENTHALPY	
PRESSURE	=
MOISTURE	=
S.G. BLOWDOWN FLOW	
F_a	
TEMPERATURE	=
MATERIAL	=
DENSITY	
PRESSURE	=
DELTA P	=
S .G. BLOWDOWN ENTHALPY	
PRESSURE	=

* Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246)

TABLE 12
SECONDARY SIDE POWER CALORIMETRIC MEASUREMENT UNCERTAINTY

COMPONENT	INSTRUMENT UNCERTAINTY	POWER UNCERTAINTY
		+a,c
FEEDWATER FLOW (HEADER)		
LEFM		
FEEDWATER DENSITY		
TEMPERATURE		
PRESSURE		
FEEDWATER ENTHALPY		
TEMPERATURE		
PRESSURE		
STEAM ENTHALPY		
PRESSURE		
MOISTURE		
NET PUMP HEAT ADDITION		
STEAM GENERATOR LOOP BLOWDOWN FLOW		
ORIFICE (FLOW COEFF.)		
THERMAL EXPANSION COEFF.		
TEMPERATURE		
MATERIAL		
DENSITY		
PRESSURE		
DELTA P		
STEAM GENERATOR BLOWDOWN ENTHALPY		
PRESSURE		
BIAS VALUES		
POWER BIAS		
2 LOOP UNCERTAINTY (WITHOUT BIAS)		
2 LOOP UNCERTAINTY (WITH BIAS VALUES)		

*, **,*** Indicates sets of dependent parameters
**** Provided by Wisconsin Electric Power and Caldon (NPL 2001-0195 & NPL 2001-0246)

IV. RESULTS/CONCLUSIONS

The preceding sections provide the methodology to account for pressure, temperature, power and RCS flow uncertainties for the RTDP analysis. The uncertainty calculations have been performed for Point Beach Units 1 & 2 with the plant specific instrumentation and calibration procedures. The following table summarizes the results and the uncertainties that are used in the Point Beach 1 & 2 safety analysis.

Parameter	Calculated Uncertainty	Uncertainty Used in Safety Analysis
Pressurizer Pressure	[+a,c	±50.0 psi (random)
Tavg		±6.0 °F (random) (includes bias)
Power (feedwater venturis) Power (LEFM on header)	]	±2.0% power (random) (at 1656 Mwt-NSSS power) ±0.6% power (random) (at 1679 Mwt-NSSS power)
RCS Flow (plant computer) (calorimetric measurement)	±2.06% flow (random) +0.25% flow (bias) [ ]+a,c	±2.4% flow (random) (includes bias)

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4. Westinghouse letter NS-TMA-1837, T. M. Anderson to S. Varga, NRC, dated 6/23/78.

5. Westinghouse letter NS-EPR-2577, E. P. Rahe Jr. to C. H. Berlinger, NRC, dated 3/31/82.

6. Westinghouse Letter NS-TMA-1835, T. M. Anderson to E. Case, NRC, dated 6/22/78.

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9. Regulatory Guide 1.105 Rev. 2, "Instrument Setpoints for Safety-Related Systems", dated 2/86.

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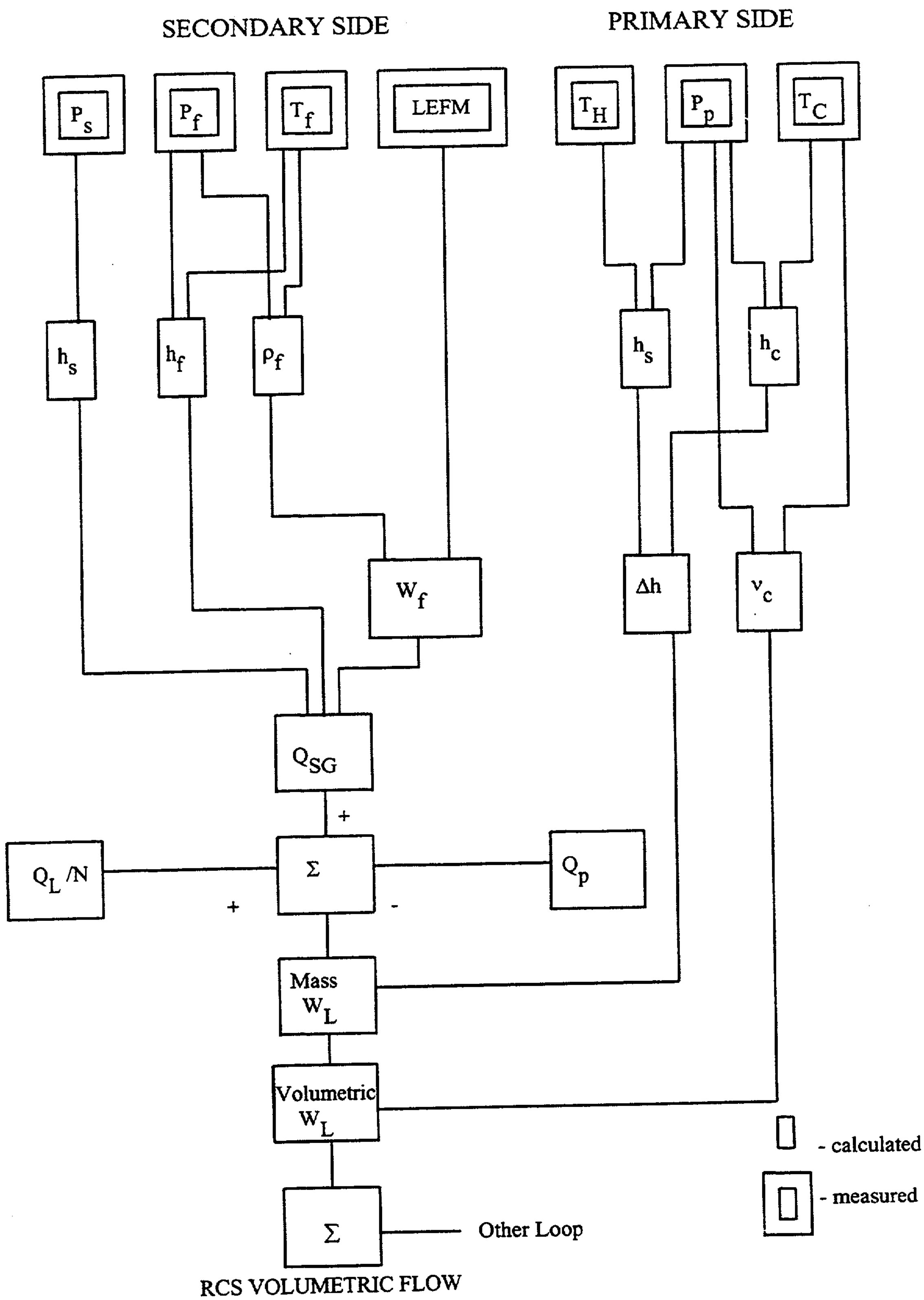

Figure 1
Calorimetric RCS Flow Measurement (Using LEFM on Feedwater Header)

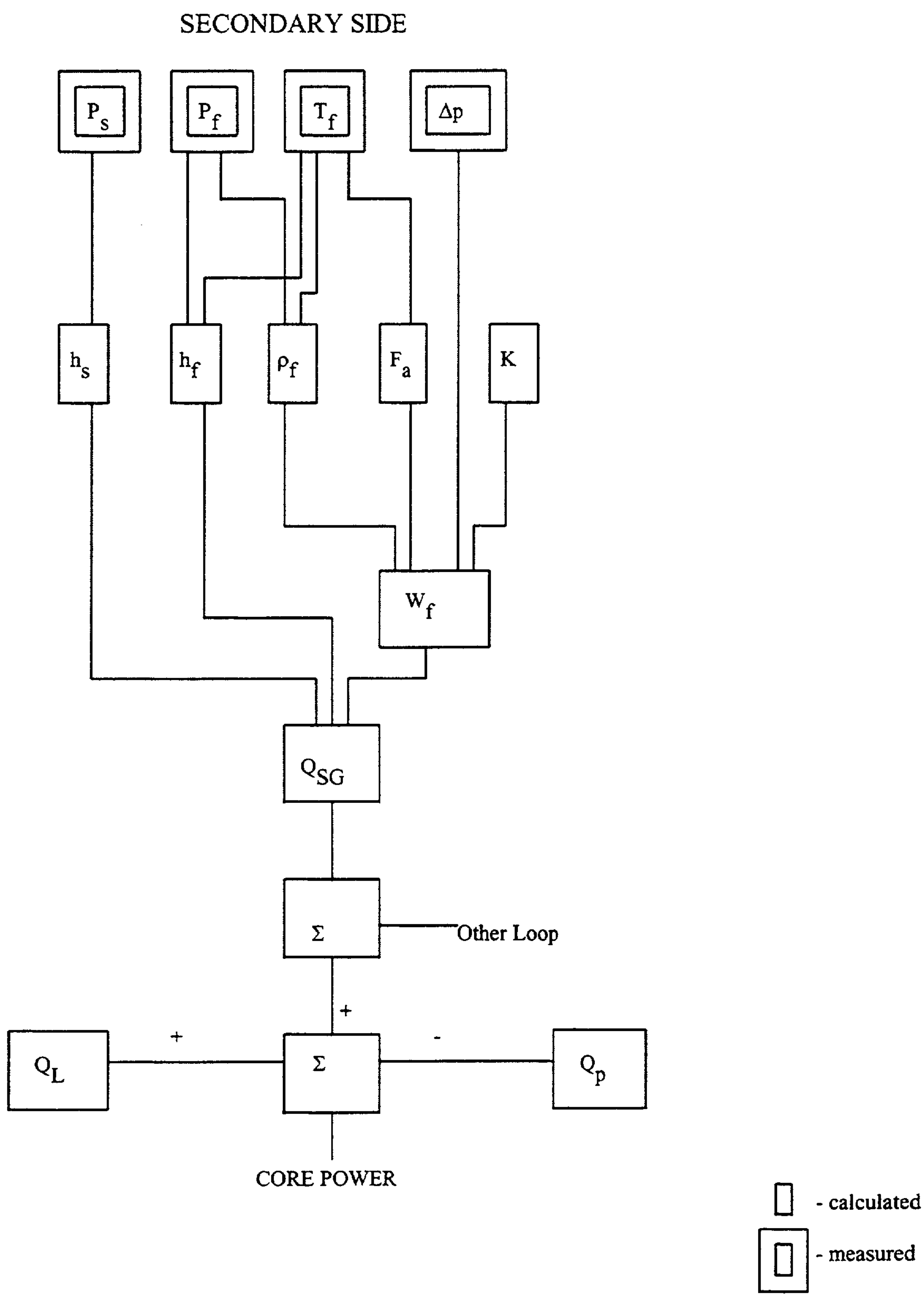

Figure 2
Calorimetric Power Measurement (Using Feedwater Venturis)

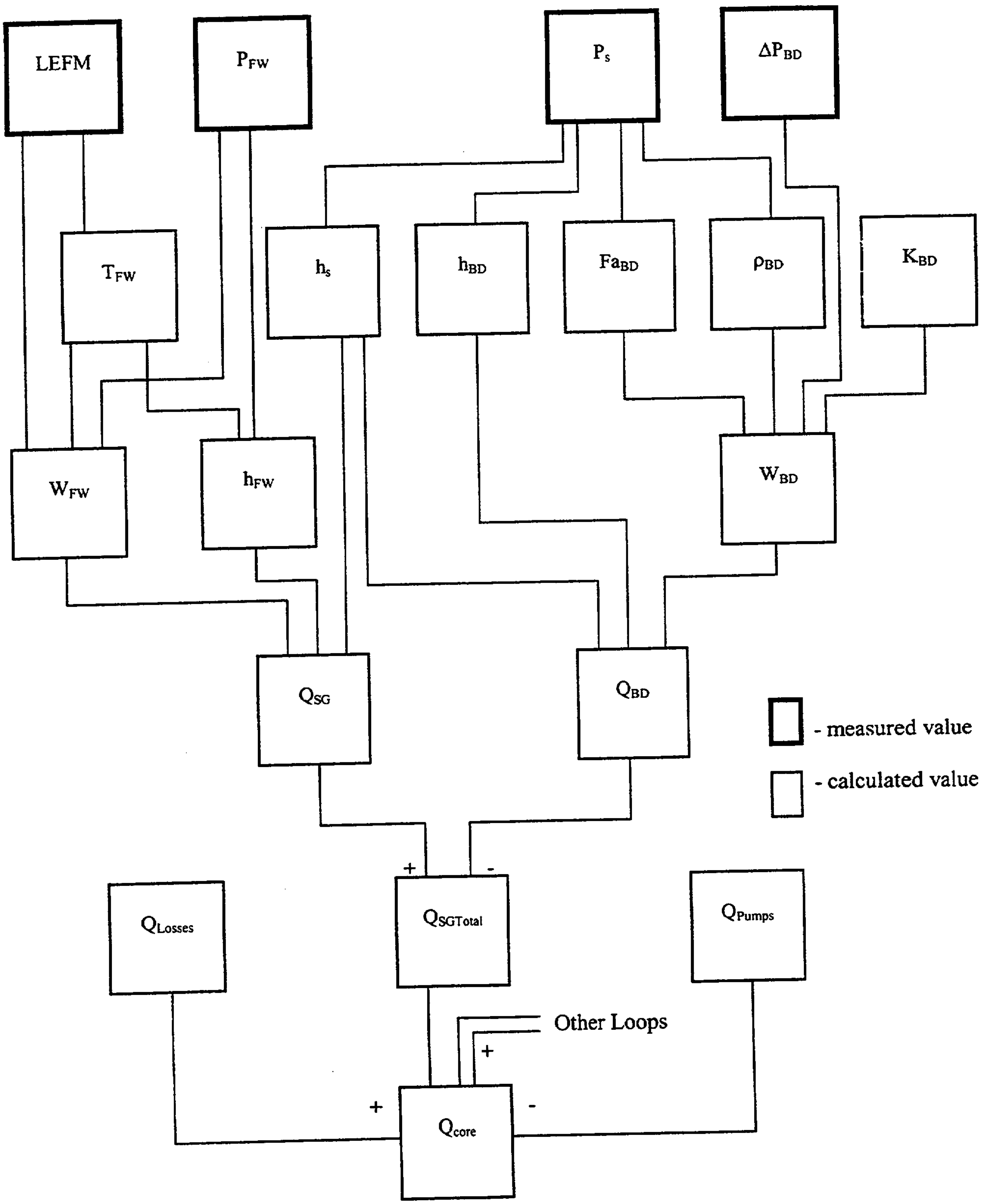

Figure 3 Calorimetric Power Measurement (using LEFM on Feedwater Header)