NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Point Beach Nuclear Plant
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NRC 2002-0075

10 CFR 50.90

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Dockets 50-266 and 50-301
Point Beach Nuclear Plant, Units 1 and 2
Responses To Requests For Additional Information
License Amendment Request 226
Measurement Uncertainty Recapture Power Uprate

By submittal dated April 30, 2002, Nuclear Management Company, LLC (NMC), submitted a request for an amendment to the Operating Licenses and Technical Specifications (TS) for Point Beach Nuclear Plant (PBNP), Units 1 and 2. The purpose of the proposed amendment was to increase licensed rated thermal power (RTP) based on a measurement uncertainty recapture (MUR) power uprate.

In a June 6, 2002, teleconference between the Nuclear Regulatory Commission (NRC) staff and PBNP plant staff, the NRC staff requested additional information in support of the proposed amendment. The NMC response to the staff's questions was submitted in letter NRC 2002-0053 dated June 26, 2002.

During conference calls between NMC representatives and NRC staff on June 27, July 9, August 6, and August 19, 2002, the NRC staff requested additional information in relation to the April 30, 2002, submittal.

Attachment 1 of this letter provides the NMC responses to the NRC staff's requests for additional information (RAIs). In support of these RAIs, Caldon, Incorporated (Caldon) and Westinghouse Electric Company (Westinghouse) have provided documentation in attachments 2 through 7. Attachment 2 contains two copies of the proprietary version of the Caldon Engineering Report MPR-1619, Appendix B, to support question 3 from the NRC I&C Branch. Attachment 3 contains two copies of the non-proprietary version of MPR-1619, Appendix B. Attachment 4 contains the Caldon authorization letter and the accompanying affidavit, CAW-02-02. Attachment 5 contains two copies of the Westinghouse proprietary report, "Enclosure 2, Point Beach Units 1 and 2, 1.4% MUR Power Uprate Support, NRC RAI Responses Set 1 Question 4." This attachment supplements question 4 from the NRC Mechanical and Civil Engineering Branch. Attachment 6 contains two copies of the non-proprietary version, "Enclosure 3, Point Beach Units 1 and 2, 1.4% MUR Power Uprate Support, NRC RAI

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Responses Set 1 Question 4." Attachment 7 contains the Westinghouse authorization letter, accompanying affidavit CAW-02-1542, Proprietary Information Notice, and Copyright Notice. Attachment 8 contains PBNP Final Safety Analysis Report (FSAR) markups for 14.1.9, "Loss of External Electrical Load," 14.1.10, "Loss of Normal Feedwater," and 14.1.11, "Loss of All AC Power to the Station Auxiliaries," which have been provided for information only in support of Question 1 from the NRC Reactor Systems Branch.

As Attachment 2 contains information proprietary to Caldon, it is supported by an affidavit (Attachment 4) signed by Caldon, the owner of the information. The affidavit sets forth the basis on which the identified proprietary 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 subject information, which is proprietary to Caldon, be withheld from public disclosure in accordance with 10 CFR 2.790. Correspondence with respect to the proprietary aspects of the Caldon information or supporting the Caldon Affidavit, should reference the appropriate authorization letter and be addressed to Calvin R. Hastings, President and CEO, Caldon, 1070 Banksville Avenue, Pittsburgh, Pennsylvania, 15216.

As Attachment 5 contains information proprietary to Westinghouse, it is supported by an affidavit (Attachment 7) 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.

No changes to the initially proposed amendment result from this additional information. Furthermore, NMC has determined that this supplement 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, NMC concludes that the proposed supplement meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that 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.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 29, 2002.

A J Cayia
Site Vice President

Attachments

1. Responses to Requests for Additional Information
2. MPR Associates, Inc. Engineering Report MPR-1619, Rev.0, Appendix B (Proprietary)
3. MPR Associates, Inc. Engineering Report MPR-1619, Rev.0, Appendix B (Non-Proprietary Version)
4. Caldon Authorization Letter and Accompanying Affidavit, CAW-02-02
5. Enclosure 2, Point Beach Units 1 and 2, 1.4% MUR Power Uprate Support, NRC RAI Responses Set 1 Question 4" (Proprietary)
6. Enclosure 3, Point Beach Units 1 and 2, 1.4% MUR Power Uprate Support, NRC RAI Responses Set 1 Question 4" (Non-Proprietary)
7. Westinghouse Authorization Letter, Accompanying Affidavit, CAW-02-1542, Copy Right Notice, and Proprietary Information Notice
8. FSAR Markups for 14.1.9, "Loss of External Electrical Load," 14.1.10, "Loss of Normal Feedwater," and 14.1.11, "Loss of All AC Power to the Station Auxiliaries" (for information only)

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

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST 226

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Attachment 1

Responses to Requests for Additional Information

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information raised during telephone conferences on June 27, July 9, August 6, and August 19, 2002. Each of the NRC staff's questions is restated below with the Nuclear Management Company, LLC (NMC) response following.

Questions from NRC Materials and Chemical Engineering Branch

1. **In Section 3.4.5 of Attachment 1 to the submittal on power uprate for the Point Beach plant, the licensee stated that although the flow accelerated corrosion (FAC) program will be affected by the power uprate, the identified changes are insignificant and are not expected to cause significant increase of inspection intervals or repairs. Also, in Section 3.6.1 of the same attachment it stated that after power uprate the flow in the feedwater and condensate lines is expected to increase by 2 percent or less. Although 2 percent increase may not significantly affect operation of these systems, its effect on FAC are difficult to predict without performing suitable analysis. The Licensee is requested, therefore, to either justify that after power uprate the loss of materials due to increase in FAC is insignificant, or to describe what preventive measures it will be using to account for this loss.**

   Response:

   Predictive analysis was performed using the CHECWORKs computer code. Based on the predictive analysis, the wear rates increased in some lines. The largest percentage increase was in a small portion of the condensate system where the typical component wear rate will increase 7.5 percent (from 10.6 mils/yr to 11.4 mils/yr). Most pre-uprate wear rates are low and small increases were predicted following the measurement uncertainty recapture (MUR) uprate (most less than 1 mils/yr). Therefore, the changes on FAC are insignificant. Additionally, wear rate assessments are a part of the FAC program and this program will remain in place following the MUR uprate.

   The system of most concern, however, is the feedwater system because it contains the most energy. The following table summarizes the change in final feedwater parameters:

CHECWORKs Parameter	***Actual Value (pre up-rate)***	***Actual Value (post up-rate)***
Operating Temperature	431.5°F	431.6°F
Velocity	16.2 ft/sec	16.4 ft/sec
Wear Rate	3.11 mils/yr	3.14 mils/yr

2. **Section 3.4 - Please provide a summary of the operational assessment for any active degradation mechanisms under power uprated conditions.**

   Response:

   The EPRI Steam Generator Inspection Guidelines define active degradation mechanisms as the combination of at least ten new indications of degradation (greater than or equal to 20 percent through-wall) and previous indications that have an active growth rate that is greater than or equal to 25 percent of the repair limit per cycle in any steam generator. The guidelines also define active degradation mechanisms as new or previously identified indications which have a one cycle growth rate equal to or exceeding the repair limit.

   With these definitions, there are no active degradation mechanisms in Point Beach Unit 1 or Unit 2 steam generators. The Point Beach steam generators will continue to be assessed for degradation per site directives that meet the EPRI guidelines.

Questions from Reactor Systems Branch

1. **Provide a statement confirming that your Chapter 14 analyses in the PBNP FSAR version (06/01) are the analyses you cite for the power uprate. If they are not, please submit the current analyses.**

   Response:

   The analyses in Tables 3.2.1-1, 3.2.2-1, and 3.2.3-1 of the MUR uprate submittal are the current analyses of record that were documented in the 06/01 revision of the Point Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR) with the exception of the analyses described below.

   FSAR 14.1.10, "Loss of Normal Feedwater"
   The analysis referenced in the MUR uprate submittal is that documented in the 06/01 revision of the FSAR as amended by an FSAR change request that will be processed in 2003. This analysis was originally performed to support the Fuel Upgrade License Amendment Request (LAR) 210 and used a reactor power of 1650 MWt and a higher auxiliary feedwater (AFW) flow. Therefore, FSAR markups submitted for information in LAR 210 cite these assumptions. Subsequent to the Fuel Upgrade and LAR 210, PBNP determined the AFW flows and the power level could not be supported and performed the analysis at 1518.5 MWt (plus 2 percent power measurement uncertainty) with lower AFW flow as well as other input assumption changes. The changes were evaluated using the site 10 CFR 50.59 process and incorporated into the 06/01 FSAR revision. After the 06/01 FSAR revision, an error in the differential pressure from the steam generators (SG) to the main steam safety valves (MSSV) during full flow conditions was identified. The analysis was reperformed using the new pressure drop and the results met all the acceptance criteria. The changes have been accepted through the site 10 CFR 50.59 process and will be included in the 2003 FSAR update. The FSAR markups have been included as Attachment 8 to this letter for information only.

FSAR 14.1.9, "Loss of External Electrical Load," and FSAR 14.1.11, "Loss of All AC Power to the Station Auxiliaries"
The analyses referenced in the MUR uprate submittal are those documented in the 06/01 revision of the FSAR. However, these accident analyses were also affected by the change in differential pressure from the SGs to the MSSVs described above. These analyses were also reperformed using the new pressure drop. The results met all acceptance criteria. The changes have been accepted through the site 10 CFR 50.59 process and will be included in the 2003 FSAR update. The FSAR markups have been included in Attachment 8 of this letter for information only.

FSAR 14.2.5.c, "Rupture of a Steam Pipe (Containment Response)"
The analysis referenced in the MUR uprate submittal is a plant specific analyses that supports LAR 223, "Containment Pressure," submitted to the NRC for review on 01/11/2002. The MUR uprate submittal states that this accident is currently under review by the NRC.

FSAR Sections 14.2.4, "Steam Generator Tube Rupture (Radiological)," and 14.2.5.B, "Rupture of a Steam Pipe (Radiological Consequences)"
The analyses referenced in the MUR uprate were those reviewed and approved by the NRC in SERs dated 07/01/1997 and 07/09/1997. These analyses were also documented in the 06/01 revision of the FSAR. The analyses were revised in late 2001 based on changes to the inputs used to calculate the equilibrium iodine appearance rate. The revised analyses were found acceptable and changed through the site 10 CFR 50.59 process. Dose rates were updated in the 06/02 FSAR revision and the FSAR changes summarized in letter NRC 2002-0057, "Annual 10 CFR 50.59 Summary Report for 2001 Point Beach Nuclear Plant, Units 1 and 2," sent to the NRC on 06/28/2002.

2. **On the first paragraph of page 33 (Att. 1) of the submittal, in the discussion of the pressure-temperature (P-T) curve fluence, it is stated that: "This [the need to revise the P-T curves] is monitored through review of the EFPY values in the monthly operating reports." Is the monitoring accomplished only through EFPY even after the uprate? Are your EFPY limits listed on Page 32 of your submittal adjusted because of the uprated power conditions? If so, what are the new limits?**

   Response:

   Applicability of Pressure-Temperature (P-T) Curves is assured through the monitoring of effective full power years (EFPY) as well as neutron fluence. The PBNP P-T curves are generated considering the limiting reactor vessel material at the reactor vessel inside surface. The reactor vessel fluence values for these limiting materials are then determined for use in adjusted reference temperature (ART) calculations and P-T curve generation. Therefore, the P-T curves are applicable to a specific fluence value.

Monitoring of P-T curves is performed through monitoring of neutron fluence as well as the projected EFPY for these fluence values. The current P-T limit curves for Point Beach Units 1 and 2 have been approved for 25.59 EFPY (Unit 1) and 30.51 EFPY (Unit 2). These operating times correspond to calculated fast ($E > 1.0$ MeV) neutron fluence levels of 2.25E+19 n/cm^2 and 2.61E+19 n/cm^2, respectively. Nuclear Management Company (NMC) has previously committed to providing an update to the P-T limit curves by October 1, 2003. At the time of new P-T curve update, Unit 1 will have reached approximately 25.34 EFPY and Unit 2 approximately 24.94 EFPY. Therefore, the P-T curve update will occur before the actual expiration dates of the current curves.

The 1.4 percent power uprate for PBNP Units 1 and 2 is scheduled to occur after November 1, 2002. At that time, Unit 1 will have accrued approximately 24.42 EFPY of operation and Unit 2 will have accrued approximately 24.02 EFPY. Therefore, between the time of the MUR uprate and the issuance of revised P-T limit curves each unit will operate for a period of 0.92 EFPY.

In the pre-uprate condition, Unit 1 accrues fluence at an average rate of 8.79E+17 n/cm^2 per EFPY. The corresponding rate for Unit 2 is 8.54E+17 n/cm^2. Thus, in the 0.92 EFPY time period between the MUR uprate and P-T limit curve update, the incremental fluence accrued without the power uprate would be 8.06E+17 n/cm^2 and 7.83E+17 n/cm^2 for Units 1 and 2, respectively. The corresponding fluence accrual with the effects of the MUR uprate included would be 8.17E+17 n/cm^2 (Unit 1) and 7.94E+17 n/cm^2 (Unit 2). Based on these comparisons, the net additional fluence to each vessel due solely to the uprate is:

<u>For Unit 1</u>:

Incremental Fluence Due to Uprate = [8.17E+17 – 8.06E+17] = 1.10E+16 n/cm^2

<u>For Unit 2</u>:

Incremental Fluence Due to Uprate = [7.94E+17 – 7.83E+17] = 1.10E+16 n/cm^2

This additional fluence accrued in the short period of time between the MUR uprate and the scheduled revision of the P-T limit curves is insignificant when compared to the total fluence that forms the basis for the current curves. Therefore, adjustment of the current expiration dates is not warranted at this time and the EFPY limits that appear on page 32 of the PBNP submittal were not adjusted due to power uprate conditions.

3. **Does the methodology used in the determination of the fluence values on page 32 of the submittal adhere to the guidance of RG 1.190?**

Response:

The fluence values used in the generation of current PBNP P-T curves were not calculated directly using Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001. This Regulatory Guide was issued following the generation of the PBNP P-T curves. However, the PBNP neutron exposure calculations followed the draft reports of Regulatory Guide 1.190 (DG-1025).

Neutron exposure calculations for the PBNP P-T curves were performed in WCAP-12794,"Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 1," Revision 4, and WCAP-12795, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 2," Revision 3. In these reports, the "calculation-only" exposure of the reactor pressure vessel was developed using absolute plant specific neutron transport calculations. The "best estimate" fluence values were not used in the determination of the PBNP P-T curves.

4. **Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and revision 1, NSAL 02-4, and NSAL 02-5, to document the problems with the Westinghouse designed steam generator (SG) water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, 2002, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG water level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affect the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how Point Beach Units 1 and 2 account for these uncertainties documented in these advisory letters in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still adequate.**

Response:

The subject NSALs have been reviewed internally for applicability to PBNP. The following discussion summarizes the results of those reviews.

Impact of NSAL Issues on the Steam Generator Water Level Trip Setpoints

NSAL 02-3 is concerned with the effect of a pressure drop across the mid-deck plate on the steam generator low-low level trip setpoint. As noted in Attachment 1 of NSAL 02-3, the estimated pressure drop of the mid-deck plate is essentially zero for PBNP. According to Westinghouse, this is due to the large flow areas that are present in the mid-deck plate for both types of steam generators (44F and delta 47). Therefore, the mid-deck plate does not result in any additional pressure drops, and no corrections are required for the safety analysis, Technical Specification minimum, or nominal plant low-low water level setpoints.

NSAL 02-4 is concerned that the uncertainty calculations do not reflect the void content of the two-phase mixture above the mid-deck plate. The concern is that the additional error introduced could impact the high steam generator water level trip function (i.e., feedwater trip).

At PBNP, the high-high steam generator water trip setpoint is set at 78 percent Narrow Range Span (NRS). The total loop error given for the SG narrow range (NR) level at 100 percent NRS is approximately +2 percent/-7.6 percent. The location of the top of the mid-deck plate is estimated to be at 87 percent NRS for Unit 1 and 85 percent NRS for Unit 2. According to the information in NSAL 02-4, the void fraction above the mid-deck plate for PBNP is 15 percent. Therefore, the maximum reliable indicated level (MRIL) corresponding to this void fraction and the mid-deck plate locations is approximately 98 percent NRS for both steam generator types (i.e., the additional error introduced due to the void fraction above the mid-deck plates is ≤2 percent NRS). The MRIL represents the indicated level if the froth level is at the upper tap, and not accounting for other instrument uncertainties. By comparing the MRIL (-2 percent) plus total loop error (-7.6 percent) with the high-high steam generator water level trip setpoint (78 percent), it can be seen that there is sufficient margin with the froth level at the elevation of the upper tap. Note that as the steam generator water level reaches and passes the high-high level trip, it is still below the elevation of the mid-deck plate, and the void fraction above the mid-deck plate does not yet apply. Therefore, the issue of the void fraction above the mid-deck plate does not impact the acceptability of the high-high steam generator trip.

NSAL 02-5 is concerned with steam generator water level control system measurement uncertainty issues. In particular, the water level uncertainties assumed as initial conditions for safety analyses may not be bounding due to: 1) neglect of the pressure drop across the mid-deck plate and 2) neglect of velocity head effects at the lower tap.

As noted in the plant applicability statement from NSAL 02-3, PBNP is listed as a plant with a zero pressure drop across the mid-deck plate. Therefore, the impact of the mid-deck plate pressure drop on the SG level measurement uncertainty is zero. In addition, the velocity head effect on the indicated steam generator water level has already been considered for PBNP during the Unit 1 Level Tap Relocation and Unit 2 Replacement Steam Generator Programs. The velocity head term is then used in the calculation that determines total loop errors. Therefore, the concerns raised in NSAL 02-5 are not applicable to PBNP.

Impact of NSAL Issues on the Safety Analyses and ATWS Events

As discussed above, the issues raised in the NSALs do not impact the trip setpoints used at PBNP and, therefore, the current analyses remain limiting. For information, the following is noted:

- The LOCA analysis assumes a trip on low pressurizer pressure and, therefore, is not affected by the SG level trips.
- The non-LOCA analyses that use the low-low steam generator level trip are the loss of AC power and loss of normal feedwater. These trip setpoints are unaffected because the mid-deck plate pressure drop is zero.
- The high-high steam generator water level trip setpoint is well below the mid-deck plate and is unaffected by the void fraction issue of NSAL 02-4.
- The ATWS Mitigating System Actuation Circuitry (AMSAC) system at PBNP does not use the steam generator water level as a trip parameter and is not affected.

5. **Your power uprate application relies on the Westinghouse generic ATWS analysis to demonstrate the acceptance of the analytical results. Provide a discussion of the ATWS analyses that are applicable to the specific plant and power uprate conditions, and justify that the assumptions for the analyses are adequate as they relate to input parameters such as the initial power level, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, RCS volume, steam generator pressure, auxiliary feedwater (AFW) flow rate and its actuation delay time, and the setpoint for the AMSAC system to actuate the AFW and trip the turbine. The submittal should include a discussion and applicable values of the unfavorable exposure time for the MTC assumed in the analyses.**

Response:

For Anticipated Transients Without Scram (ATWS), operation of the PBNP units at the current licensed power level is supported by the Westinghouse generic ATWS analyses that were submitted to the NRC in Westinghouse letter NS-TMA-2182 dated December 31, 1979. Consistent with the guidelines prescribed in NUREG-0460, these ATWS analyses were performed assuming initial operating conditions consistent with nominal plant conditions. In the analyses presented in NS-TMA-2182, a nominal reactor power of 1520 MWt was assumed for the reference 2-Loop Westinghouse pressurized water reactor (PWR) configuration applicable to the PBNP units. As prescribed by NUREG-0460, the ATWS analyses presented in NS-TMA-2182 also included sensitivity analyses for variations in specific parameters. These sensitivity analyses included 2 percent uncertainties in power levels.

Two-Loop Plant ATWS Analyses

The Loss of (External) Load ATWS and the complete Loss of Normal Feedwater ATWS are the most limiting events for reactor coolant system (RCS) pressure, but the peak RCS pressures remain less than 3200 psig. This 3200 psig pressure corresponds to a conservative bound for the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion.

The reference ATWS analyses presented in NS-TMA-2182 included various Westinghouse PWR configurations applicable at the time. These LOFTRAN analyses included 2, 3, and 4-loop PWRs with various steam generator models. The reference plant used in the analyses was a 4-loop Westinghouse PWR with Model 51 steam generators. For the Point Beach units, the generic ATWS analyses applicable at that time are those for a 2-loop PWR with Model 44 steam generators and a core power of 1520 MWt. The Point Beach units are currently licensed with a core power of 1518.5 MWt. Unit 1 has Westinghouse Model 44F steam generators and Unit 2 has Westinghouse Delta 47 steam generators. Relative to ATWS events, the performance characteristics of these steam generators closely resemble those of the Model 44 steam generators modeled in the generic ATWS analyses.

A comparison of key plant parameters important to ATWS analyses for Point Beach Units 1 and 2 and the reference 2-loop plant with Model 44 steam generators modeled in NS-TMA-2182 is given in the following table:

Comparison of Point Beach to Reference 2-Loop/Model 44 Input Parameters

Parameter	Point Beach Units	Reference 2-Loop Plant (Reference Case)
Reactor Power (MWt)	1518.5 (current)	1520
Relief Capacity (PORV and RCS Safety Valve, Number, Capacities, and Setpoints)	Identical to reference plant	Identical to Point Beach units
RCS Volume (ft^3)	6400 (Unit 1) 6548 (Unit 2)	6230
Steam Generator Pressure (psia)	Design Pressure: 1100 Nominal Operating Pressure: 777/752 (0%/10% Tube Plugging)	Design Pressure: 1100 Nominal Operating Pressure: 750
Auxiliary Feedwater (AFW) Flowrate	> 400 gpm[1]	800 gpm
AFW Initiation	60 seconds after AMSAC signal.	60 seconds after AMSAC signal.
AMSAC (ATWS Mitigating System Actuation Circuitry) Setpoint	PBNP AMSAC not intiated by process variable setpoint.[2]	Analyses assume AMSAC signal initiates in 60 seconds

[1] Point Beach Critical Safety Procedure CSP-S.1, "Response to Nuclear Power Generation/ATWS," contains a continuous action statement that directs the operator to maintain total (auxiliary) feed flow greater than 400 gpm until level greater than 29 percent in at least one steam generator. The reference analysis considered a worst-case single failure of a turbine-driven AFW pump that results in a 50 percent reduction in AFW flow as a sensitivity study.

[2] AMSAC actuates when it determines that all main feedwater pumps have tripped, or when main feedwater flow to the steam generators has been blocked due to valve closures.

Loss of Load ATWS

The Loss of Load ATWS analysis for the reference plant resulted in a peak system pressure of 2974 psia. Transient results for 3-loop and 2-loop plants with 51 series steam generators gave lower peak RCS pressures of 2861 psia and 2753 psia, respectively (note that the core power level for the 2-loop plant with Model 51 steam generators was 1650 MWt). The transient results of a loss of load ATWS for Model 44 steam generators were similar to the Model 51 results. The peak RCS pressures calculated were 2979 psia for the 4-loop plant, 2839 psia for the 3-loop plant, and 2753 psia for the 2-loop plant. These results demonstrate that the RCS pressure results for Loss of Load ATWS calculated for a 4-loop plant with Model 51 steam generators are bounding for the 2-loop plant with Model 44 steam generators. This is important because input parameter sensitivity studies were performed using the 4-loop plant with Model 51 steam generators.

The sensitivity of the peak RCS pressure for changes in input parameters relative to the reference case for the Loss of Load ATWS are given in Table 5.1-2 of the NS-TMA-2182 report. The changes in peak pressure for plant parameters that differ significantly from the reference case (4-loop, Model 51 steam generators) are as follows:

Parameter Sensitivity Case	Change in Maximum RCS Pressure Relative to Reference Case (psia)
One-Half Auxiliary Feedwater Flow	+64
RCS Volume + 10%	+42
Reactor Power + 2%	+44

As discussed above, the ATWS results for a 4-loop, Model 51 steam generator plant bound the 2-loop plant with Model 44 steam generators. Given the baseline Loss of Load peak RCS pressure of 2753 psia for a 2-loop plant with Model 44 steam generators, and the bounding sensitivities to peak pressure given above (2753 + 64 + 42 + 44 = 2903 psia), the margin to 3200 psia is approximately 297 psi. Therefore, it can be concluded that the Loss of Load ATWS RCS peak-pressure analysis for Point Beach with a 1.4 percent uprate would remain below 3200 psia.

Loss of Normal Feedwater ATWS

The Loss of Normal Feedwater ATWS analysis for the generic reference plant resulted in a peak system pressure of 2848 psia. Transient results for 3-loop and 2-loop plants with 51 series steam generators gave lower peak RCS pressures of 2783 psia and 2753 psia, respectively (note that the core power level for the 2-loop plant with Model 51 steam generators was 1650 MWt). The transient results of a Loss of Normal Feedwater ATWS for Model 44 steam generators were similar to the Model 51 results. The peak RCS pressures calculated were 2857 psia for the 4-loop plant and 2717 psia for the 3-loop plant with Model 44 steam generators. No results were given for the 2-loop plant in the NS-TMA-2182 report; the 3-loop plant results will be used here for the Loss of Normal Feedwater ATWS peak RCS pressure. Thus the peak pressure results for the 4-loop plant increased by 9 psi, while the peak pressure for the 3-loop case decrease by 66 psi. Given the small change in peak pressure for the 4-loop, Model 44 steam generator plant, and the trend for lower peak pressure for the 3-loop plant case, it is concluded that the RCS pressure results for Loss of Normal Feedwater ATWS calculated for a 4-loop plant with Model 51 steam generators are bounding for the 2-loop plant with Model 44 steam generators.

The sensitivity of the peak RCS pressure for changes in input parameters relative to the reference case for the Loss of Normal Feedwater ATWS are given in Table 5.2-2 of the NS-TMA-2182 report. The changes in peak pressure for plant parameters that differ significantly from the reference case are as follows:

Parameter Sensitivity Case	Change in Maximum RCS Pressure Relative to Reference Case (psia)
One-half Auxiliary Feedwater Flow	+31
RCS Volume + 10%	+18
Reactor Power + 2%	+23

Given the baseline peak RCS pressure of 2717 psia, and the bounding sensitivities of the change in maximum RCS pressure given above (i.e., 2717 + 31 +18+ 23 = 2789 psia), the margin to 3200 psia is approximately 411 psi. Therefore, it can be concluded that the Loss of Normal Feedwater ATWS peak-pressure analysis for Point Beach with a 1.4 percent uprate will remain below 3200 psia.

AMSAC Setpoint

The AMSAC for Point Beach follows the "Logic 3" design described in WCAP-10858P-A. This AMSAC design was accepted generically by the NRC for Westinghouse plants. The Logic 3 circuitry actuates when it determines that all main feedwater pumps have tripped, or when main feedwater flow to the steam generators has been blocked due to valve closures. Since this AMSAC design anticipates the plant response due to loss of main feedwater pumps prior to the reactor protection system (RPS) detecting an anticipated operational occurrence, the AMSAC actuation is delayed to allow the RPS to function as designed. The Point Beach AMSAC employs a fixed delay time of 30 seconds.

Unfavorable Exposure Time

From a Westinghouse/WOG perspective, the term "Unfavorable Exposure Time" or UET represents the duration of a given fuel cycle, for a specific plant configuration, in which the total core reactivity feedback is insufficient to preclude exceeding a peak RCS pressure of 3200 psig following an ATWS event. UET was defined by the Westinghouse Owners Group (WOG) for use in a more detailed ATWS probability risk assessment (PRA) model developed as part of the "Westinghouse/WOG ATWS Rule Administration Process," WCAP-11992. (See also WOG Comment 1, USNRC, "Final Report Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," page C-13.) The concept of UET is also being applied in a revised Risk-Informed ATWS PRA model supporting an ongoing WOG ATWS PRA program.

To determine the UET values used in this ATWS PRA model, the reactivity feedback required to just yield a peak RCS pressure of 3200 psig is first determined by specific ATWS transient analyses. The reactivity feedback conditions for a given reload core model are then compared to the transient reactivity feedback models to determine the value of UET for a given plant configuration. For the Westinghouse/WOG ATWS PRA model, a total of 12 UET values are typically determined and used. The term UET as defined and applied above was not a term directly used in the basis of the Final ATWS Rule as documented in SECY-83-293. In SECY-83-293, the terms "favorable MTC" and "unfavorable MTC" are applied in the discussion of the simplified ATWS PRA model. These terms are not the same as UET defined above.

No UETs have been calculated for Point Beach for this power uprate. However, favorable/unfavorable MTCs for ATWS for Point Beach are discussed in the following section.

Moderator Temperature Coefficients

As discussed above, the Point Beach licensing basis analysis for ATWS is contained in NS-TMA-2182, which is also the analysis basis for the AMSAC design described in WCAP-10858P-A. As indicated in NS-TMA-2182, the Point Beach units are considered "Alternative 3" plants as defined in NUREG-0460, Volume 3. In NS-TMA-2182, the Alternative 3 plant assumptions include a MTC valid for 95 percent of core life. That is, the favorable MTC will be more negative than -8 pcm/^{o}F for 95 percent of the time that the core power is greater than 80 percent of nominal.

The MTC values for the current operating cycles at Point Beach Units 1 and 2 were reviewed. For Unit 1 Cycle 27, the predicted MTC is more negative than –8 pcm /°F for 95 percent of the cycle burnup (95 percent favorable MTC). For Unit 2 Cycle 26, the predicted MTC is more negative than –8 pcm /°F for 99 percent of the cycle burnup. The MTC values assume Hot Full Power (HFP), with All Rods Out (ARO) and Equilibrium Xenon (EQXE) in the core. The difference between the two cores is attributed to Unit 1 having one reload of 422 Vantage+ fuel (422V+, 0.422 inch diameter rods) and roughly two-thirds of the core of Optimized Fuel Assembly (OFA) fuel (0.400 inch diameter rods), while Unit 2 Cycle 26 has had two reloads of 422V+ fuel. The difference in the 422V+ fuel rod diameters reduces the volume of water between the fuel rods and should be a benefit in terms of meeting the ATWS Favorable MTC criteria.

Both Point Beach units are operated as base load units and power reductions are infrequent. The purpose of downpowers is for testing or for maintenance, and not for load follow. Reactivity is controlled by adding or removing boron, and the control rod position is maintained just out of the core. Therefore, the MTC information provided above based on hot full power, all rods out, and equilibrium xenon reflects the current and future operation of Point Beach Units 1 and 2.

Conclusion

The results of the ATWS base analyses and sensitivity analyses which support the current AMSAC design as presented in NS-TMA-2182 support operation at a maximum reactor power of 1550 MWt (102 percent of 1520 MWt) for the 2-Loop PWR plant configuration. For the subject PBNP Unit 1 and 2 power uprate, the maximum reactor power level, including uncertainty, remains unchanged at 1549 MWt (102 percent of 1518.5 MWt or 100.6 percent of 1540 MWt). The 1.4 percent power uprate is achieved by reducing the uncertainty on power from 2 percent to 0.6 percent and maintaining the same net maximum reactor power with uncertainty. With the 1.4 percent power uprate, the licensed reactor power level increases from 1518.5 MWt to 1540 MWt. This power level and the Point Beach plant design remains bounded by the reactor power and other sensitivities investigated in the generic ATWS analyses in NS-TMA-2182.

6. **Upon reviewing LBLOCA models for power uprates, the NRC recently found plants that require changes to their operating procedures because of inadequate hot leg switch-over times and boron precipitation modeling. Discuss how your analyses account for boric acid buildup during long term core cooling and discuss how your predicted time to initiate hot leg injection corresponds to the times in your operating procedures.**

Response:

PBNP is an Upper Plenum Injection (UPI) plant, in which the low head Emergency Core Cooling System (ECCS) pumps (Residual Heat Removal or RHR) deliver flow to core deluge nozzles directly to the upper plenum. PBNP also has the capability to inject high head Safety Injection (SI) flow into the same core deluge nozzles (although this requires manual realignment). As such, the hot leg switchover procedure that is applied at some Westinghouse plants does not apply at PBNP. The ECCS fluid is continuously injected into the upper plenum directly above the core as long as the low head pumps are in use and the RCS pressure is low enough. This is analogous to a plant with hot leg injection capability transferring ECCS flow from cold leg to hot leg injection at some specified time after a Loss of Coolant Accident (LOCA).

The core deluge valves to the UPI nozzles open on an SI signal to allow flow from the low-head SI (RHR) pumps. To ensure that the flow path to the UPI nozzles is available, the emergency operating procedure for transfer to recirculation contains a verification step to ensure that the core deluge valves are open. Since the core deluge valves open on an SI signal, there will be flushing and mixing of the upper plenum and upper core area after the RCS depressurizes below the RHR shutoff head. Therefore, boron precipitation during the long term cooling phase of the accident should not occur. The B train high head SI injection line is also capable of providing core deluge for higher RCS pressures.

It is noted that this issue has been addressed in a previous NRC Safety Evaluation Report dated December 24, 1975. This was a review of response to a December 27, 1974, Atomic Energy Commission Order for Modification of License implementing the requirements of 10 CFR 50.46. It was the position of PBNP staff that boron precipitation would not occur due to the unique UPI design. However, the SER identified an acceptable time to perform a "hot leg switchover" of 14 hours to assure boron solubility. The establishment of core deluge occurs much earlier than that time frame, and is confirmed when transfer to sump recirculation is performed. Typically, the transfer to sump recirculation after a large break LOCA occurs well before 14 hours. Therefore, core deluge is also verified well before the 14 hour transfer time stated in our earlier submittal.

7. **For the power level cited in your LOCA models, demonstrate that your models and procedures continue to comply with 10 CFR 50.46 during the switch-over from the RWST to the Containment Sump.**

   Response:

   Section 9 of the Fuel Upgrade LAR (LAR 210) submitted discussion of new LOCA analyses that included use of the best estimate LOCA (BELOCA) methodology, VANTAGE + (422V+) fuel design, and ZIRLO cladding. The analysis was performed for an uprated core power rating of 1650 MWt, and includes an allowable uncertainty of 2 percent. The results of the analyses demonstrate that the five acceptance criteria in 10 CFR 50.46 are met (Section 9.3 of LAR 210), including long-term cooling. This conclusion is based on the conditions at the end of the WCOBRA/TRAC calculations, which indicate that the transition to long term cooling is underway even before the entire core is quenched.

8. **For your Uncontrolled RCCA Withdrawal at Power accident analysis, the fuel centerline temperatures are not mentioned. Describe how the analyses at 1650 MWt continue to meet your fuel centerline temperature limits.**

   Response:

   The fuel centerline temperature limit is met for the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power event by ensuring that the peak core average power (heat flux) does not exceed a prescribed limit (118 percent of rated thermal power) for a wide range of reactivity insertion rates, for several different power levels, and for both minimum and maximum reactivity feedback. The linear heat generation rate (kW/ft) that corresponds to the prescribed heat flux limit is confirmed to be less than the kW/ft value that would cause fuel centerline melting. As the 118 percent limit was met for the spectrum of Uncontrolled RCCA Withdrawal at Power cases analyzed, the fuel centerline temperature limit was met.

9. **For your CVCS Malfunction accident at power, your old accident analysis indicated that you had 17.9 minutes for operator action to prevent a return to criticality, and your new analysis indicated that this value is 18.2 minutes, when using the Vantage 422+ fuel at an increased power level of 1650 MWt. However, when using the new fuel assemblies, you decrease the available RCS volume. Additionally, you increased the reactivity insertion rate in your accident analysis. Both of these parameters should indicate a decrease in operator action time to prevent criticality. Explain what analysis parameters changed between your two analyses to increase the available operator action time, and explain why these assumptions continue to bound the current RCS design.**

   Response:

   The most significant difference between the two analyses is the assumed least negative boron worth value, which is used to calculate a minimum reactivity insertion rate. With the minimum reactivity insertion rate, a maximum time for generating either a high neutron flux or overtemperature ΔT signal for reactor trip is determined (based on the Uncontrolled RCCA at Power transient analysis results) and factored into the total time available for operator action. The least negative boron worth values assumed in the previous and new analyses are -5 pcm/ppm and –7 pcm/ppm, respectively, which are confirmed to be conservative. These translate into reactivity insertion rates of 0.83 pcm/sec and 1.17 pcm/sec, respectively. Using the Uncontrolled RCCA at Power transient analysis results, these reactivity insertion rates correspond to reactor trip times of 74.2 seconds and 52.6 seconds, respectively. The difference in the reactor trip times (21.6 seconds (0.36 minutes)) translates into more operator action time after reactor trip in the new analysis. This more than compensates for the effect of the reduced reactor coolant system volume associated with the 422V+ fuel. In summary, margin in the available operator action time is gained by assuming a less conservative, least negative boron worth value.

10. **For the loss of external electrical load transient at 102% power, your FSAR indicates that you use 2200 psia for the starting point when modeling the pressurizer pressure (without pressure control). Given that you operate at 2250 psia and that your uncertainty is + 50 psi, explain how the 2200 psia assumption is more limiting than a 2300 psia for the overpressure condition. Also, if you use 2300 psia as the starting condition, what peak pressure is achieved? Is this pressure below 110% of your design values?**

    Response:

    The Loss of External Electrical Load (LOL) transient assumes an initial pressurizer pressure of 2200 psia (2250 psia nominal minus 50 psia uncertainty). The lower pressure is used to delay the time to reactor trip on high pressurizer pressure. A delay to reactor trip increases the amount of energy produced at 102 percent of full power that must be removed from the RCS, and results in a higher peak RCS pressure. Consequently, there is no case run for an initial pressurizer pressure of 2300 psia.

11. **In comparison to your previously approved accident analyses, your current Loss of Normal Feedwater accident, Table 14.1.10-1 indicates a substantial decrease in the time to reach the Low-Low steam generator water level trip. Subsequently, the remaining event times are significantly changed. Describe the assumptions that changed and the reasons why these assumptions are still bounding for your current accident analysis.**

Response:

In the 06/27/02 conference call it was recognized that the NRC reviewer was referring to FSAR markups transmitted with the Fuel Upgrade (LAR 210). These markups used the 06/98 revision of the FSAR. Table 14.1.10-1 of the 06/98 FSAR revision inferred the AFW delay to be 10 minutes (600 seconds) while the text of FSAR Section 14.1.10 stated the delay was 5 minutes (300 seconds). The calculation supporting the LONF analysis correctly assumed 5 minutes (300 seconds). Although the correct AFW delay time of 5 minutes was assumed in the calculation, it was incorrectly transcribed to FSAR Table 14.1.10-1 as 600 seconds. Therefore, there was a 300 second (600 – 300 = 300 seconds) editorial error in the FSAR event times following the low-low steam generator water level trip. This error was identified and corrected in the 06/99 revision of the FSAR, sent to the NRC on 06/23/99. The LAR 210, submitted on 06/22/99, contained markups using the 06/98 FSAR. The 06/98 FSAR revision was used for the LAR 210 submittal based on the timing of the submittal, internal review requirements, and the 06/99 FSAR revision schedule. Based on the 300 second error discussed above, the event sequence time changes noted in FSAR markups included in LAR 210 appear more significant then they actually were.

In the 06/01 FSAR revision, there was a change made under 10 CFR 50.59 to the analysis assumption for the reactor trip on low-low steam generator water level. The trip was increased from 10 percent NRS to 17 percent NRS. This has the effect of decreasing the amount of time to reactor trip, and reducing the total amount of power produced between the initiation of the transient and the reactor trip. This trip setpoint is still conservative for PBNP since the low-low level trip setpoint uncertainty is < 3 percent NRS, the Technical Specification minimum trip setpoint is 20 percent NRS, and the plant trip setpoint is 25 percent NRS. The 06/01 analysis also changed the feedwater flow assumption to 200 gpm and the power level assumption to 1518.5 MWt (core) plus two percent uncertainty, which was different than those assumptions in LAR 210.

Attachment 8 contains the FSAR markups for the current analysis of record for the LONF analysis to be incorporated in the 2003 FSAR revision. The acceptance criterion for this transient is that the pressurizer does not overfill. Based on the assumptions used in the current analysis, the peak pressurizer volume remains below the acceptance criteria value for the maximum pressurizer volume. The analysis remains bounding for the 1.4 percent power uprate since it assumes a two percent power uncertainty.

12. **Additionally, in comparison to your previously approved accident analyses, your current Loss of All AC Power to Station Auxiliaries transient, Table 14.1.11-1 indicates a substantial decrease in the time to reach the Low-Low steam generator water level trip. Subsequently, the remaining event times are significantly changed. Describe the assumptions that changed and the reasons why these assumptions are still bounding for your current accident analysis.**

Response:

As with Question 11 above, the Loss of All AC Power (LOAC) FSAR markups transmitted with the Fuel Upgrade (LAR 210) contained the 300 second error in the AFW delay. The calculation supporting the LOAC analysis correctly assumed 5 minutes (300 seconds), but the time was incorrectly transcribed in FSAR Table 14.1.11-1 as 600 seconds. Therefore, there was a 300 second (600 –300 second) editorial error in the FSAR event times following the low-low steam generator water level trip. This error was identified and corrected in the 06/99 revision of the FSAR, sent to the NRC on 06/23/99. The LAR 210, submitted on 06/22/99, contained markups using the 06/98 FSAR. The 06/98 FSAR revision was used for the LAR 210 submittal based on the timing of the submittal, internal review requirements, and the 06/99 FSAR revision schedule. Based on the 300 second error discussed above, the event sequence time changes noted in FSAR markups for Table 14.1.11-1 included in LAR 210 appear more significant then they actually were.

Attachment 8 contains the FSAR markups for the current analysis of record for the LOAC analysis to be incorporated in the 2003 FSAR revision. The acceptance criterion for this transient is that the pressurizer does not overfill. Based on the assumptions used in the current analysis, the peak pressurizer volume remains below the acceptance criteria value for the maximum pressurizer volume. The analysis remains bounding for the 1.4 percent power uprate since it assumes a two percent power uncertainty.

13. **For the Loss of AC Power to Station Auxiliaries accident, what is the most limiting single failure, and when taken, does this event still meet the DNBR and pressure acceptance criteria? What are the peak pressures and DNBR values reached?**

Response:

The most limiting single failure for the LOAC transient is the failure of a turbine driven AFW pump. The turbine driven AFW pump can provide 400 gpm to the two steam generators on its respective unit. Failure of this pump will leave two motor driven AFW pumps to deliver 200 gpm each to two units (or four steam generators). In the case of a LOAC transient, it is assumed that 200 gpm is delivered to one steam generator. The results of this analysis demonstrate the acceptance criterion is met, which is that the pressurizer does not overfill.

Note that the LOAC and LONF events are not specifically analyzed as the limiting Departure from Nucleate Boiling Ratio (DNBR) or RCS pressure transients. The Loss of External Electrical Load (LOL) transient is bounding for the LONF and LOAC with respect to DNBR and RCS pressure criteria. Therefore, the FSAR mark-ups provided in the fuel upgrade license amendment request (LAR 210) did not contain results for DNBR or RCS peak pressure. However, the peak RCS and steam generator pressures calculated for the LOAC transient cases are 2420 psia and 1190 psia, respectively. DNBR results were not calculated since the pressurizer overfill analysis is a non-RTDP (Revised Thermal Design Procedure) analysis.

14. **For the last approved Loss of Normal Feedwater accident analysis, the FSAR stated that one motor driven AFW pump delivers 200 gpm of flow to one steam generator. In your latest FSAR update, however, it states that one motor driven AFW pump provides 200 gpm (split flow) to two steam generators. However, your FSAR figure 10.2-1, sheet 1 indicates that your motor driven pumps are not capable of supplying split flow between two steam generators in the same unit. Explain how the motor driven AFW pumps are capable of supplying the 200 gpm split flow and how the split flow assumption continues to be conservative versus the non-split flow case. Additionally, you assume that one motor driven AFW pump fails to deliver flow to the steam generators. Given this condition and given your assumption that the motor driven AFW pumps are capable of split flow, could you lose all motor driven AFW pump flow because of your assumed single failure?**

Response:

The statement above that the motor driven pumps are not capable of split flow between two steam generators in the same unit is correct. The PNBP AFW system uses a steam turbine-driven pump for each unit. Each turbine driven pump is capable of supplying 400 gpm to a unit and can split the flow to the two steam generators in that unit. The two motor driven pumps are common to both units. Each pump has a 200 gpm capacity with the A pump supplying both units' A SGs and the B pump supplying both units' B SGs (100 gpm each unit). For further description of the system, please see PBNP FSAR Section 10.2, "Auxiliary Feedwater System (AF)."

The assumption for AFW flow described in the 06/01 revision of FSAR Section 14.1.10 is misleading in that one Motor Driven Auxiliary Feedwater (MDAFW) pump cannot physically supply flow to two SGs in the same unit. The actual assumed AFW flow for the LONF transient is described below. A clarification to the AFW assumption on this page has been prepared for the 2003 FSAR revision.

Westinghouse sensitivity analyses indicate it is slightly more conservative to assume AFW flow to two SGs for the LONF transient. Therefore, the assumption of 100 gpm flow to each SG was used for the LONF analysis case. Delivery of at least 100 gpm to each steam generator can be provided by a combination of either: 1) two motor-driven AFW pumps or 2) one turbine driven AFW pump. The limiting single failure for this scenario is the loss of one turbine driven AFW pump, which leaves two motor driven AFW pumps to deliver at least 100 gpm to each steam generator (each pump is capable of delivering 200 gpm). Therefore, the assumption of 100 gpm AFW flow using both the motor driven AFW pumps is conservative.

15. **Similarly, for the Loss of all AC Power transient, the previous analysis stated that one motor driven AFW pump delivers 200 gpm to one steam generator, and your current analysis states that the AFW system delivers 200 gpm to one steam generator. Describe the differences between these two assumptions and confirm that your analysis continues to be conservative given single failure.**

Response:

Westinghouse sensitivity analyses indicate it is slightly more conservative to assume an AFW flow of 200 gpm to one steam generator for the LOAC transient. The difference in the assumption is the AFW system must be able to supply the required flow, not just one motor-driven pump. In actuality, this assumption does not change since the flow of 200 gpm can be provided by: 1) one motor driven AFW pump or 2) one turbine driven pump.

The limiting single failure for this scenario is the loss of a turbine driven AFW pump. This leaves one motor driven pump to deliver 200 gpm to one steam generator, and does not credit the availability of the second motor driven AFW pump. Therefore, the single failure assumption for the AFW system is conservative.

16. **For the Natural Circulation Cooldown event, you state that the, "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." Have these uncertainty values been reviewed and approved by the NRC? If so, please provide an appropriate reference. If not, please provide the analyses to support your bounding conclusion.**

Response:

Although it is unclear whether the ANSI/ANS-5.1-1979 standard was formally reviewed and accepted by the NRC, it has been used in regulatory processes as stated in the forward to the standard. Additionally, the application of the ANSI/ANS-5.1-1979 decay heat standard and the two-sigma (2σ) uncertainty has been reviewed by the staff and found acceptable for use at PBNP. This review is documented in SER dated 07/09/97 regarding the decay heat values used in for the containment pressure and temperature analysis. The standard and 2σ uncertainty are also referenced in the PBNPs FSAR Sections 14.1.10, "Loss of Normal

Feedwater," 14.1.11, "Loss of All AC Power to the Station Auxiliaries," and 14.3.4, "Containment Integrity Evaluation." The natural circulation cooldown calculation referenced in the MUR uprate submittal was not reviewed by the NRC. The following paragraphs describe the uncertainty applied to the decay heat values used in the subject calculation and disposition the additional margin on the decay heats to encompass the MUR uprate.

A review of Table 4 of ANSI/ANS-5.1-1979 shows that for all times $\geq$ eight (8) seconds after reactor shutdown, a bounding value for one standard deviation (1σ) uncertainty in decay heat is two percent. A 2σ uncertainty then corresponds to four (4) percent for all values of $t \geq$ eight seconds.

The decay heats in the natural circulation cooldown calculation are based on a minimum time after reactor shutdown of two minutes. Therefore, a $\geq$ eight seconds, 2σ uncertainty of four percent is applicable to the decay heats. The decay heats assumed were calculated using the ANSI/ANS-5.1-1979 standard. The document transmitting the decay heats stated that the 2σ uncertainty needed to be applied to these decay heats and then incorrectly stated that 2σ uncertainty was equivalent to eight percent. The eight percent was subsequently applied to the decay heats in the natural circulation cooldown calculation. Therefore, there was a four percent conservative margin on the assumed decay heats in addition to the 2σ uncertainty of ANSI/ANS-5.1-1979.

The additional four percent of margin applied to the decay heats can be used to offset the power increase of the MUR uprate. The decay heat values were determined using ANSI/ANS-5.1-1979, Section 3.6, "Simplified Method for Determining Decay Heat Power and Uncertainty." Using this method, reactor core decay heat is linear with respect to the maximum reactor thermal power level. This relationship is shown in equation 12 of ANSI/ANS-5.1-1979. Therefore, a percent increase in power would result in the same percent increase in the decay heat values. The additional four percent margin on the decay heat inputs to the calculation can be was used to disposition the increased decay heats of the MUR uprate. A formal addendum to the calculation documents the additional conservatism of the input decay heats and uses two percent of the additional margin to disposition the additional decay heat for the MUR uprate. Application of the predicted increase in decay heats would still provide an effective two percent additional margin in the computed reactor core decay heat values used in the natural circulation cooldown calculation. Therefore, the calculation results were unchanged and remain conservative for a MUR uprate.

17. **Your submittals indicate that you are using the same computer codes for your 1650 MWt accident analyses as you used for your previous analyses (1518.5 MWt). Confirm, for the power level assumed in your accident analyses, that you continue to meet all of the limitations and conditions spelled out in the NRC SERs accepting the codes and methodologies.**

    Response:

    The PBNP power uprate analyses and evaluations were performed using currently approved analytical techniques to demonstrate compliance with the licensing criteria and standards that apply to PBNP. The codes have been controlled and verified under the Westinghouse configuration control and verification. The codes used in the supporting accident analysis were used in accordance with the applicable limitations and restrictions at the assumed power level.

18. **For the Dropped RCCA transient, you state that the transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394. Please provide the reference for the NRC's approval of this methodology. If it has not been approved, describe why it acceptable for licensing applications.**

    Response:

    This methodology is approved as documented in WCAP-11394-P-A. The Westinghouse Owners Group submitted this WCAP to the NRC on May 22, 1987. It was accepted for referencing in license applications by letter to the Westinghouse Owners Group Chairman on October 23, 1989.

19. **For the Loss of External Electrical Load Transient, your analyses at the 1650 MWt condition for the pressurizer control case significantly changed the order of events and their timing. Describe the changes made to the model and how these changes continue conservatively bound the 1650 MWt transient case.**

    Response:

    Notable input differences between the two analyses, which can affect the order of events and their timing, are provided as follows:

Changes to Loss of External Electrical Load

	Old Analysis Value	New Analysis Value
Core Power	1518.5 MWt	1650 MWt
Reactor Vessel Flow Rate	181,800 gpm	182,400 gpm
Core Bypass Flow Fraction	5.5%	5.3%
Initial Pressurizer Pressure	2000 psia	2250 psia
Overtemperature ΔT reactor trip setpoint gains: K1 K2 K3	 1.23 0.022 0.001	 1.255 0.0149 0.00072
Pressurizer PORV Setpoint	2069.6 psia	2350 psia

All new analysis inputs, which are mostly attributed to the fuel transition, have been confirmed to be applicable (conservative) for the 1650 MWt case.

20. **Your rupture of a Steam Pipe accident analyses at 1650 MWt indicate a significant change to the modeling of the Core Boron Concentration levels versus time from the previous analyses. Describe the changes made to the model and how these changes continue conservatively bound the 1650 MWt transient case.**

    Response:

    Modeling borated water in the passive, safety injection accumulators was the change that affected the core boron concentration level. No credit was taken for the passive accumulators in the previous analysis. An accumulator boron concentration of 2000 ppm was credited, which is conservative with respect to the minimum value required by the Technical Specifications (2600 ppm). The analysis remains conservative with this modeling change.

21. **FSAR Section 14.1.6, "Reduction in Feedwater Enthalpy Incident," indicates that the accident analyses was performed at 1518.5 MWt and that it was not reanalyzed for the 1650 MWt conditions. Additionally, FSAR section 14.1.7, "Excessive Load Increase Incident," indicates that the analysis was not performed for the 1650 MWt power level, but a generic statepoint analysis shows that a 1650 MWt power level is acceptable. This condition implies that the power level in the analysis did not change. And finally, FSAR Table 14.0-1 indicates that your Loss of Normal Feedwater accident is analyzed for 102% of 1650 MWt, whereas FSAR section 14.1.10 implies that the core power is 102% of 1518.5 MWt. These power levels tend to disagree with the values listed in your submittal dated April 30, 2002, Table 3.2.1-1. For these accidents, please clarify your current power level assumptions.**

Response:

FSAR Section 14.1.6, "Reduction in Feedwater Enthalpy Incident," was originally analyzed at 1518.5 MWt. During the Fuel Upgrade/Power Uprate Program, this analysis was not reanalyzed but evaluated by Westinghouse. It is documented in the reload transition safety report from Westinghouse that the decrease in feedwater temperature incident is similar to and usually less severe than the Excessive Load Increase Incident of FSAR Section 14.1.7. Therefore, a hand calculation was performed to show that the change in feedwater temperature resulting from the opening of the feedwater bypass valve was bounded by the change that would be required to create the 10 percent increase in steam load for the Excessive Load Increase. By showing this in the hand calculation, and demonstrating acceptable DNBR results for the Excessive Load Increase Incident, the DNB design basis and other applicable acceptance criteria are met for this event. Therefore, this event was not reanalyzed for 1650 MWt but has been proven to be bounding for 1650 MWt. Table 3.2.1-1 of the submittal shows the current FSAR power assumption to be 1650 MWt because this incident has been evaluated and remains bounded at 1650 MWt.

FSAR Section 14.1.7, "Excessive Load Increase Incident," was originally analyzed at 1518.5 MWt. During the Fuel Upgrade/Power Uprate Program, this analysis was not reanalyzed but a simplified state point calculation performed because, historically, this event does not lead to a serious challenge to the acceptance criteria and the reactor trip is not generated. Limiting state points were generated and the results confirmed that core DNB limits are not challenged following this event. The historical text describing the previous analysis was left in the FSAR for information and the conclusions from the statepoint calculation were added, stating that this event has been evaluated at 1650 MWt and that acceptance criteria are met. Table 3.2.1-1 correctly states 1650 MWt, the maximum power level that has been evaluated at and found acceptable for this accident analysis.

FSAR Section 14.1.10, "Loss of Normal Feedwater," is currently analyzed at 102 percent of 1518.5 MWt. The FSAR Table 14.0-1 lists the power level incorrectly for this analysis. During the Fuel Upgrade/Power Uprating Program, the LONF accident was originally performed at uprated power of 1650 MWt. FSAR sections were marked up appropriately. Subsequent to the fuel upgrade amendment NRC SER, review of the AFW system resulted in revised input assumptions for this analysis including the current power level of 1518.5 MWt. The latter analysis became the correct analysis of record and FSAR Section 14.1.10 was again revised. A footnote was added to FSAR Table 14.0-2 to identify the power level difference, but FSAR Table 14.0-1 was not updated correctly for the Loss of Normal Feedwater. The error in this table was identified during the MUR submittal preparation and has been incorporated into the FSAR change request for the MUR uprate, which is expected to go in the 2003 FSAR revision. Table 3.2.1-1 of the submittal correctly states 102 percent of 1518.5 MWt for the Loss of Normal Feedwater analysis. This analysis has been further updated since the 06/01 FSAR revision as stated in Question 1 (also see Attachment 8) and continues to use the 1518.5 MWt power assumption.

Questions from NRC I&C Branch

1. **Attachment 2 and 3 included with the letter submitted WCAP-14787 and WCAP-14788 revision 1, which discussed the Westinghouse Revised Thermal Design Procedure (RTDP) instrument uncertainty methodology. These topical reports reference the ISA Standard S67.04, 1994 and RG 1.105 rev 2. However, the document fails to provide a positive statement that the methodology meets the requirements identified in the standard and RG. Also, explain why the uncertainty PEA, RCA, RMTE, CA, and CMTE are 0.00.**

   Response:

   WCAP-14787 and 14788 meet the requirements identified in ISA Standard S67.04, 1994, and RG 1.105, Revision 2.

   The uncertainties questioned above are from Table 1 of WCAP-14787 (page 8) and are explained below:

   PEA – There are no orifices or venturis in this instrument channel. Therefore, the PEA term is set to 0.0.

   RCA, RMTE - Since a loop calibration is performed from the input of the process instrumentation to the control board indicator at Point Beach, the loop calibration accuracy is included in the RCA_{IND} term and the RCA term is set to 0.0. The measurement and test equipment accuracy for the loop calibration is included in the $RMTE_{IND}$ term and the RMTE term is set to 0.0.

   CA, CMTE – There is no calibration of the controller module that compares the reference pressure signal to the measured pressure signal. Therefore, the CA term is set to 0.0. This is a closed-loop control system and the plant operators adjust the control system reference pressure signal until the desired pressure is obtained on the control board indicators. The measurement and test equipment accuracy for the controller module calibration is the $CMTE_{IND}$ term and the CMTE term is set to 0.0.

2. **Section 3.1.5 discusses that in the event of the unavailability of LEFM the power will be reduced to 1520 MWt rather than 1518.5 MWt, which is the original power. This is justified based on the fact that the calculated venturi's uncertainty is 1.87 as opposed to 2 percent required by the Appendix K rule. However, attachment 2 lists the venturi's uncertainty on page 26 different from 1.87. Clarify this discrepancy.**

Response:

Page 26 of WCAP-14787, Revision 1, reported a power measurement uncertainty of ±1.87 percent with a +0.04 percent bias for power measurement using the feedwater venturis. The positive 0.04 percent bias added to the power measurement uncertainty would result in instrumentation indicating a higher power level than what was actually being produced. This is not an operating concern since the licensed Rated Thermal Power (RTP) is not exceeded. However, a negative uncertainty is of concern since the actual reactor thermal output (RTO) could be higher than the indicated RTP, and therefore, could exceed the licensed RTP. Since the bias was positive, it was not used to determine the RTP that could be operated at with the feedwater venturis.

However, during review of this question, Westinghouse determined that the positive bias reported on page 26 of WCAP-14787 would actually apply in both the positive and negative direction. The bias of concern is the result of a Rosemount feedwater flow transmitter vibration penalty. This bias should have been expressed as a limit of error with either a positive or negative value. The 0.04 percent bias can no longer be applied in only the positive direction. Based on Westinghouse's conclusion, the limiting uncertainty associated with the feedwater venturis is now ±1.91 percent.

The relaxation of the Appendix K rule allows PBNP to use this uncertainty as opposed to the 2 percent power measurement uncertainty of the original Appendix K rule. Applying the 1.91 percent power measurement uncertainty allows for a 0.09 percent uprate from the current RTP of 1518.5 MWt to 1520 MWt. This power level (1520 MWt) with the 1.91 percent power measurement uncertainty results in the same initial power level of 1549 MWt for the accident analysis. The appropriate power level will be used in the PBNP Technical Requirements Manual (TRM), procedures, and operator training. Additionally, the error in the bias will be corrected in a revision of the WCAP-14787.

3. **Section 3.1.4 states that Point Beach LEFM spool pieces were installed in the early 1980's by Westinghouse and were not calibrated in a site-specific hydraulic geometry prior to installation. Section 3.1.4 further states that the licensee has used the statistical approach discussed in Caldon report ER-80P that has been approved by the staff. Explain how the information needed for the statistical approach was received. Also, is there a copy available of MPR Associates Report, MPR-1619, dated May 1995, which is used to determine the hydraulic profile factor for Point Beach.**

Response:

The PBNP Unit 1 and 2 spool pieces were not calibrated in a site specific hydraulic geometry when they were installed in the early 1980s. In 1995, Caldon subcontracted MPR Associates (MPR) to perform feedwater flow measurement uncertainty calculations for the Leading Edge Flow Meter (LEFM) 8300 measurement system that Caldon was installing at PBNP Units 1 and 2. The Caldon system used the previously installed Westinghouse spool pieces. These calculations required the spool piece hydraulic profile factor and the profile factor's factor's uncertainty. Since site specific calibration was not performed, MPR had to statistically develop the profile factors and the profile factor uncertainties for the spool pieces.

As at Texas Utilities' Comanche Peak Steam Electric Station (which received NRC MUR uprate approval on September 30, 1999), the PBNP Unit 1 and Unit 2 profile factors are based on prior Alden Research Laboratories testing of eight Westinghouse LEFM spool pieces that established the profile factor and its uncertainty for fully developed flow. A detailed explanation of developing profile factors and bounding uncertainty for spool pieces that have not been calibrated at a laboratory has been reviewed by the NRC and docketed on April 26, 1999, in Question 19 of the meeting summary from September 29, 1998, "Summary Of Meeting With Texas Utilities Electric Company Regarding Appendix K Exemption Request And Topical Report That Supports The Exemption Request."

The PBNP profile factors are also based on MPR's analysis of the impact of the plant specific piping geometry. In-situ path specific velocity data, collected during the commissioning of the PBNP Unit 1 and 2 LEFM 8300 flow meters, was used to validate the results of the MPR analysis. The flow meter tests providing the velocity data were conducted at the PBNP in April 1995. The basis for the PBNP profile factors and the associated uncertainty is described in detail in Appendix B of report MPR-1619, which is included in Attachment 2 of this letter. Additional supporting information about profile factors can be found in the full report from the September 29, 1998, meeting referenced above.

The statistical approach used in MPR-1619 is the same approached discussed in Caldon Topical Report, ER-80P. Caldon has reviewed and accepted the 1995 MPR report for use in the plant specific feedwater flow measurement uncertainty analysis performed for the new LEFM✓ system at PBNP. Therefore, the hydraulic profile factor and its associated uncertainty developed in the MPR-1619 report, dated May 1995, remains valid for the new LEFM✓ system.

<u>Questions from NRC Civil and Mechanical Branch</u>

1. **In Section 3.4, Attachment 2 of the reference, Table 3.4.1 provides comparison of design parameters, including core power, NSSS power, reactor coolant systems (RCS) pressure, Tave range, thermal design flow rate, steam generator (SG) tube plugging, steam pressure, steam temperature, and steam flow rate, for the current power level of 1518.5 megawatts thermal (MWt), the proposed power level of 1540 MWt and a power level of 1650 MWt which was assumed for the Unit 2 steam generator replacement project. Please provide similar data for other key design parameters including reactor inlet temperature (T cold), reactor outlet temperature (T hot), steam generator outlet temperature, feedwater temperature, and feedwater flow rate.**

Response:

The original Table 3.4.1 is reproduced here along with the additionally requested information for easier comparison.

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/loop	89,000 gpm/loop	85,200 gpm/loop
SG tube plugging	0 to 10 percent	0 to 10 percent	0 to 25 percent
Steam Pressure	612 - 857 psia	664 - 800 psia	612 - 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	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
Reactor Inlet T_{cold}	526.0 - 551.3 °F	528.0 - 544.5 °F	526 - 546.1 °F
Reactor Outlet T_{hot}	588 - 611.3 °F	588.1 - 603.5 °F	592.9 - 611.3 °F
Steam Generator T_{out} (Steam Side)	488.7 - 526.2 °F	497.3 - 518.2 °F	485.5 - 518.7 °F
Feedwater Temperature	431.5 °F	431.6 °F	440.7 °F
FW Flow Rate (assuming no blowdown)	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

2. **In reference to Section 3.4.2, the licensee stated that the impact of power uprate on reactor vessel components susceptibility to flow induced vibration (FIV) was evaluated during RSG project as documented in WCAP-14459, "Reactor Pressure Vessel and Internal System Evaluations for the Point Beach Units 1 and 2 Power Uprating/Replacement Steam Generator," 1999. Confirm whether this report has been previously reviewed and approved by the staff with regard to effects of flow-induced vibration for the power uprate condition. If not, provide detailed FIV evaluation including the methodology, assumptions, and analysis results such as the elastic-fluid stability factor, calculated vibration stresses due to steady state flow, turbulence and vortex flow, and acceptance criteria. If the detailed FIV analysis is covered by WCAP-14459, please provide the supporting documentation. Also provide a quantitative evaluation for the potential of flow induced vibration for the SG U-Bend tubes based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure (unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.**

Response Part 1, Reactor Vessel Flow Induced Vibration (FIV):

Westinghouse performed a FIV evaluation of the reactor components as part of the power uprating/Replacement Steam Generator (RSG) program (1650 MWt). The report was not previously submitted by PBNP to the NRC. Rather than submit the WCAP, which contained the entire internals evaluation, the following detailed summary of the FIV evaluation is provided.

<ins>Introduction</ins>

Flow induced vibrations of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies is to assure the structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale-model tests, as well as tests in fabricators' shops and bench tests of components, along with various analytical investigations. The results of these scale-model and in-plant tests indicate that the vibrational behavior of two-, three-, and four-loop plants is essentially similar, and the results obtained from each of the tests compliment one another and make possible a better understanding of the flow-induced vibration phenomena.

Elastic-fluid stability for the reactor internals is not a concern for the structural system since it is rigidly supported at both ends and the flow from the inlet nozzle flows downstream into the downcomer region. This has been verified and substantiated during the scale model test, in-plant tests, and the hot functional test.

The FIV response of reactor internal components depends upon reactor vessel inlet flow rates, reactor vessel inlet temperature and reactor vessel outlet temperature. The response of the lower internals (core barrel assembly) depends on the vessel inlet temperature and the inlet flow rates. The response of the upper internals (guide tubes and upper support columns) depends on the vessel outlet temperature and the flow exiting through the outlet nozzles.

Input Parameters

The Nuclear Steam Supply System (NSSS) parameters for the power uprating/RSG program (1650 MWt) were used in the evaluation of the FIV response.

Acceptance Criteria

The acceptance criteria for the FIV response are that the displacement amplitudes, strain, and the resulting stresses from the FIV evaluation remain within the endurance limit of the material for high cycle fatigue of the ASME Code Section III, Appendices 1989 Edition.

Evaluation and Results

Mechanical measurements were made at a 2-loop reference plant as part of a broad effort to obtain vibration information on the first of a kind of the two loop plants. Abundant technical data have been published on the 2-loop internals vibration assurance program; strain gage measurements, acceleration measurements, and results of static and shaker tests on guide tubes along with analytical investigations. The Westinghouse method to confirm acceptable results for the power uprating/RSG program was to perform a comparison of PBNP physical plant characteristics and the plant uprating/RSG NSSS parameters to the analytical data on FIV from the reference plant.

The hot functional test of the 2-loop reference plant was conducted at flow rates of 105,000 gpm/loop which is approximately 4 percent higher than the mechanical design flow rates of 101,200 gpm/loop for the PBNP units. The PBNP units and the reference plant have essentially the same reactor vessel geometry such as length, diameter, thickness, and support locations. The physical design of internals is essentially identical and any differences will not affect the internals response with regard to flow induced vibrations.

Since equal dynamic properties such as natural frequencies assure equal response under equal excitations, there will be no difference in the response of PBNP and the reference plant.

Lower Internals Response

Results from the scale model and in-plant tests indicate that the primary cause of lower internals' excitations is the flow turbulence generated by the expansion and turning of the flow at the transition from the inlet nozzle to the barrel-vessel annulus, and the wall turbulence generated in the down-comer. The test results from the 2-loop reference plant indicate that the vibration amplitudes and the corresponding stresses of the lower internals are extremely small.

The response of PBNP lower internals can possibly be influenced by changes in design performance capability parameters for the uprating/RSG program, such as flow rate and operating temperatures. The impact of these factors on the response of the lower internals for the reference plant has been evaluated and the results showed that there was negligible impact on the response. Since the mechanical design flows remain unaffected with the proposed changes, and since the reference plant hot functional test was performed at flow rates higher (i.e., 105,000 gpm/loop) than the mechanical design flow rates for PBNP (i.e., 101,200 gpm/loop), the existing test results are conservative and remain applicable for both PBNP Units. As mentioned previously, the other parameter that could influence the FIV response is the core inlet temperature. This temperature change due to the uprating RSG parameters implies a change in water density. This change in water density is considered to have a negligible impact on the FIV response for the uprating/RSG program.

Upper Internals Response

The significant flow induced forces on upper internals are due to random turbulences generated by the cross flows that converge on the outlet nozzles. Therefore, the guide tubes and the upper support columns which lie near the outlet nozzles will experience the maximum flow induced forces. The magnitude of these forces (fluctuating as well as steady state drag) are proportional to square of the fluid velocity at the outlet nozzles. The flow induced vibration loads on the guide tubes and the upper support columns remain essentially unchanged. Previous FIV analyses on the guide tubes and the upper support columns along with the test results show that there exist sufficient margins.

Summary

The flow induced vibration amplitudes, strains and consequently stresses are very small and well below the ASME code allowable fatigue curve. Note also that the ASME code (Section III, Appendices 1989 Edition) allowable fatigue curve is conservative. Based on the results of the analysis performed for the Uprating/RSG program in 1996, the structural integrity of the reactor internals remains acceptable with regards to flow-induced vibrations.

Response Part 2, SG FIV and U-Bend Fatigue:

Flow Induced Tube Vibration and U-Bend Fatigue– Model 44F

An evaluation of the potential for high cycle fatigue of unsupported U-bend tubes was performed. One of the prerequisites for high cycle U-bend fatigue is the formation of a dented support condition at the upper plate. This support condition is a result of a build up of corrosion products associated with drilled holes in carbon steel tube support plates (TSPs). Since the broached stainless steel support plate is designed to inhibit the introduction of corrosion products, the support condition necessary for the development of high cycle fatigue cannot occur. As a result, high cycle fatigue associated with unsupported inner row tubes also cannot occur in this model SG.

Fatigue usage associated with general FIV resulting from the most limiting uprated operating condition has also been calculated for the Unit 1 steam generators. This was performed by modifying results obtained from the original FIV report to account for the revised power level. In the original analysis it was determined that the limiting tube had a maximum FIV induced tube bending stress of <0.3 ksi. Conservative calculations indicate that when this tube is in operation at the uprated condition the corresponding maximum stress level would be approximately 0.4 ksi. This level of stress is still well below the endurance limit of ~23.7 ksi at 1E11 cycles, hence the fatigue usage factor associated with FIV induced loadings while in the uprated operating condition is 0.0.

With respect to tube wear, the baseline analysis for the Unit 1 Model 44F steam generators indicates that wear as a result of tube vibration is very small over the projected design life of the unit. The rate of tube wear resulting from the proposed uprate has been determined to increase after the uprate, but not significantly. The maximum wear has been projected to increase from ~3 mils (original pre-Uprate condition) to less than ~4 mils (post-uprate). With a typical plugging limit of 40 percent through-wall, (20 mils for a wall thickness of 0.050 inch), sufficient margin exists to account for the wear anticipated due to vibration subsequent to the implementation of the uprate.

In addition, recently obtained eddy current test (ECT) data substantiates the original analysis that there is little wear on the tubes as a result of vibration. The latest degradation assessment for Unit 1, concludes that there are no active degradation mechanisms in the Unit 1 steam generators with only a total of three tubes plugged due to anti-vibration bars (AVB) wear over the life of the steam generators, none since 1995. Only 16 active tubes show any AVB wear at all over this period with average wear of less than 1.5 percent through-wall per EFPY.

From the above, it has been determined that the limiting aspects of flow induced vibration associated with steam generator tubing, (fatigue and tube wear) will not be affected by the proposed uprate.

<u>Flow-Induced Tube Vibration and U-Bend Fatigue– Model Delta-47</u>

An evaluation of the potential for high cycle fatigue of unsupported U-bend tubes was performed. One of the prerequisites for high cycle U-bend fatigue is the formation of a dented support condition at the upper plate. This support condition is a result of a build up of corrosion products associated with drilled holes in carbon steel TSPs. Since the broached stainless steel support plate is designed to inhibit the introduction of corrosion products, the support condition necessary for the development of high cycle fatigue cannot occur. As a result, high cycle fatigue associated with unsupported inner row tubes also cannot occur in this model SG.

Fatigue usage associated with general FIV resulting from the most limiting uprated operating condition has also been calculated for the Unit 2 steam generators. This was performed by modifying results obtained from the original FIV report to account for the revised power level. In the original analysis it was determined that the limiting tube had a maximum FIV induced tube bending stress of <0.5 ksi. Conservative calculations indicate that when this tube is in operation at the uprated condition the corresponding maximum stress level would be approximately 0.7 ksi. This level of stress is still well below the endurance limit of ~23.7 ksi at 1E11 cycles, hence the fatigue usage factor associated with FIV induced loadings while in the uprated operating condition is 0.0.

With respect to tube wear, the baseline analysis for the Unit 2 Model Delta-47 steam generators indicates that wear as a result of tube vibration is very small over the projected design life of the unit. The rate of tube wear resulting from the proposed uprate has been determined to increase after the uprate, but not significantly. The maximum wear has been projected to increase from ~2.6 mils (original pre-Uprate condition) to less than ~5 mils (post-uprate). With a typical plugging limit of 40 percent through-wall, (20 mils for a wall thickness of 0.050"), sufficient margin exists to account for the wear anticipated due to vibration subsequent to the implementation of the uprate.

Actual ECT data substantiates that there is little if any wear on the tubes due to vibration. The latest degradation assessments for Unit 2 concludes that there are zero tubes in the Unit 2 steam generators with AVB wear. Therefore, the increase due to the proposed uprate is not projected to result in an unacceptable rate of wear since there is currently no measurable wear occurring in the steam generator tubes due to vibration.

From the above, it has been determined that the limiting aspects of flow induced vibration associated with steam generator tubing (fatigue and tube wear), will not be affected by the proposed uprate.

3. **In reference to Section 3.4.1, the licensee indicated that NSSS components and system analyses were performed for both 1518.5 MWt and the power level of 1650 MWt during the Unit 2 replacement steam generator (RSG) project. The licensee also indicated that the RSG analyses for Unit 2 are applicable for both PBNP units for the uprated power conditions. Provide the technical basis to justify that the analyses performed for Unit 2 with Model Delta 47 RSGs, are applicable to Unit 1 where steam generators are of Model 44F.**

Response:

To clarify the text of the submittal, the Unit 2 RSG project evaluated components and systems at 1650 MWt. The 1650 MWt condition bounds the conditions for a power level of 1518.5 MWt. The statement that the analysis performed during the Unit 2 RSG was applicable to both units was meant to clarify that the Unit 1 components and systems were also covered by this report even though the title only refers to Unit 2. The statement was not intended to mean that one SG model's evaluation covered the other model. The RSG project separately evaluated the Delta 47 and the 44F SGs and the text of the report separately documents each evaluation.

4. **In reference to Section 3.4.2, the licensee indicated that the existing analyses of record were performed during the Unit 2 Replacement Steam Generator (RSG) Project and remains valid for the proposed 1.4 percent power uprate. Table 3.4.2-1 summarizes assessments of components (including the reactor vessel, reactor core support structure and vessel internals, control rod drive mechanism, RCS piping, other NSSS piping systems, the pressurizer, the reactor coolant pump, the steam generators and their supports) by referring to RSG analysis. The licensee concluded that stresses and cumulative usage factor (CUF) for the proposed 1.4 percent power uprate at 1540 MWt remain below allowable limits based on the RSG analyses, which were not reviewed by the staff. Provide a summary description of analyses for Units 1 and 2 power uprate condition, including the methodology, computer codes, modeling, assumptions and loading combinations used in the evaluations of the above NSSS components.**

   Also provide the calculated maximum stresses and CUFs at critical locations in each component for the power uprate conditions. Identify the allowable code limits, the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

   Response:

   The PBNP submittal contains Table 3.4.2-1, "Bounding AOR for NSSS Components," that lists the various components that needed to be addressed in the MUR program. Several of these components were addressed by Westinghouse during the replacement steam generator project on Unit 2, while others were validated by a different consultant. The response to this question is separated into two separate sections: A Westinghouse response for the NSSS components and RCS piping and a response for the Other NSSS Fluid System Piping.

   <u>NSSS Components and RCS Piping</u>

   Westinghouse has prepared a response that addresses the NSSS components listed below. The response is contained within Attachments 5 and 6, proprietary and non-proprietary versions, respectively.

   - Reactor Vessel and Internals
   - Control Rod Drive Mechanisms
   - Reactor Coolant Loop Piping and Supports
   - Reactor Coolant Pumps and Motor
   - Steam Generators
   - Pressurizer
   - NSSS Auxiliary Equipment

Other NSSS Fluid System Piping

The piping and supports of the Chemical and Volume Control, Residual Heat Removal, Safety Injection and Containment Spray, Sampling, and the Component Cooling systems were reviewed to justify acceptability of these systems at the uprated power level of 1650 MWt. The review consisted of a comparison of the system design parameters for the 1650 MWt uprate to the input parameters for the current piping analysis reports (IEB 79-14). For the systems listed above, all uprated parameters of interest (temperatures, pressure, support loads, etc.) remain within the design inputs of the current analysis. Based on this, the pipe support loads also did not change. The code, code editions, and limit did not change. Therefore, the piping and supports of these systems remain acceptable at an uprated power of 1650 MWt.

5. **In Table 3.4.2-1, it is unclear whether the Balance-of-Plant (BOP) piping and supports (including main steam, condensate and feedwater, auxiliary feedwater and steam generator blowdown systems piping) were specifically evaluated for a power level of 1650 MWt. If these evaluations were performed, provide a summary of the evaluation and the calculated maximum stresses at critical locations of each affected BOP system piping and supports for the power uprate condition, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If these evaluations were not performed, provide a justification.**

   Response:

   The piping of the Main Steam, Condensate and Feedwater, Auxiliary Feedwater, and the Steam Generator Blowdown systems were reviewed to justify acceptability of these systems at the uprated power level of 1650 MWt. The review consisted of a comparison of the system design parameters for the 1650 MWt uprate to the input parameters for the current piping analysis reports (IEB 79-14). For the systems listed above, all uprated parameters of interest (temperatures, pressure, support loads, etc.) remain within the design inputs of the current analysis. Based on this, the pipe support loads also did not change. The code, code editions, and limit did not change. Therefore, the piping and supports of these systems remain acceptable at an uprated power of 1650 MWt.

6. **Provide a summary of evaluation of the effect the RSG/power uprate will have on the motor-operated valves (MOVs) program at PBNP for Generic Letter (GL) 89-10. Discuss how the safety-related MOVs at PBNP will be capable of performing their intended function(s) at the RSG/power uprate conditions. Also, discuss the effects of the RSG/power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segment for GL 96-06. Identify mechanical components, if any, whose functionality was reevaluated at the RSG/power uprate conditions.**

Response:

Generic Letter 89-10

Generic Letter (GL) 89-10 addressed concerns related to the reliable operation of MOVs. PBNP Design and Installation Guideline, DG-M17, "Guidance for the Preparation of Motor-Operated Valve (MOV) Differential Pressure Calculations," Rev. 0, January 11, 1994, specifies the method for calculation of the maximum design basis differential pressure ("worst-case scenario") expected for normal and emergency operation of MOVs. In no case did the maximum operating design system pressure change as a result of the uprate, therefore, the safety-related MOVs at PBNP will continue to be capable of performing their intended functions at MUR uprated conditions.

Generic Letter 95-07

Generic Letter 95-07 addressed concerns that pressure locking and thermal binding could render redundant safety systems incapable of performing their intended functions. Evaluations for pressure locking and thermal binding were performed at PBNP using maximum differential pressures and maximum temperatures. The maximum pressures and temperatures were taken from system design conditions and accident analysis assumptions. The maximum design conditions and accident analysis assumptions are not changing for the MUR uprate. Therefore, current evaluations for pressure locking and thermal binding remain bounding for the MUR uprate and all valves in the scope of 95-07 continue to be capable of performing their intended function.

Additionally, the NRC SER for the GL 95-07 (dated 01/08/98) stated that affected valves were either modified or procedures modified to assurance that the conditions of concern were adequately addressed. Subsequent to the SER, additional valve modifications were performed eliminating the thermal binding and pressure locking issues in several valves. There are no changes to SER conclusions based on the MUR uprate.

Generic Letter 96-06

The Generic Letter 96-06 addressed concerns that thermally induced over pressurization of isolated water filled piping section in containment. The PBNP is using its current containment integrity analysis for the MUR uprate because it was performed at 1518.5 MWt with two percent uncertainty. The pressure and temperature accident values remain as documented in the analysis of record and are still bounded. Therefore, there is no increase in the possibility of over pressurization of isolated segments of safety related piping inside containment.

<u>**Additional Questions in 08/06/02 Conference Call**</u>

1. **Heatup and Cooldown Curves: Provide estimated Peak Neutron Fluence values at the ID surface and 1/4T for limiting components at EOL for uprated conditions.**

   Response:

   The fluence values, and the corresponding EFPY applicability dates, referenced in NMC Letter 2002-0030 are for the current PBNP P-T Limits. The limiting fluence values for the current P-T curves are again listed in the following table. In addition, this table contains the projected fluence values for the End of Life (EOL) and the EOL one-quarter thickness (1/4T) locations.

   Table: Fluence Values

	Current P-T Curves	**EOL (34 EFPY)** Note 1	**EOL (1/4T)** Note 2
Unit 1	2.25×10^{19} n/cm^2	2.73×10^{19} n/cm^2	1.848×10^{19} n/cm^2
Unit 2	2.606×10^{19} n/cm^2	2.72×10^{19} n/cm^2	1.842×10^{19} n/cm^2

   Note 1- The values corresponds to the fluence at the inside surface of the limiting RPV component.
   Note 2- The 1/4T fluence values are determined using equation (3) of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials": $f = f_{surf}\,(e^{-0.24x})$ where f_{surf} is expressed in units of E19n/cm^2 (E>1MeV) and x is the desired depth in inches into the vessel wall. 6.5 inches used as vessel wall thickness.

2. **Upper Shelf Energies: Provide the predicted USE values for both Units at EOL using uprated conditions. If the predicted values are less than 50 ft-lbs and the fracture mechanics evaluation was used to demonstrate acceptable equivalent margins against fracture, indicated whether the evaluation has been reviewed and approved by the NRC staff.**

   Response:

   A fracture mechanics evaluation has been performed to examine the PBNP upper shelf energy (USE) values in limiting welds. This evaluation examined the USE values for both EOL as well as End of Life Extension (EOLE) conditions. These values were predicted to fall below the NRC (10 CFR 50) Appendix G requirements of 50 lb/ft and are listed below:

   Unit 1 EOL - 593 lb/in (49.4 lb/ft)
   Unit 2 EOL - 609 lb/in (50.8 lb/ft)

   Therefore, a fracture mechanics evaluation was performed to demonstrate acceptable equivalent margins of safety against fracture. This plant specific evaluation is contained in BAW-2255, "Effect of Power Upgrade on Low Upper-Shelf Toughness Issue," dated May 30, 1995. The staff has not reviewed this plant specific evaluation.

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST 226

MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Attachment 7

Westinghouse Authorization Letter, Accompanying Affidavit, CAW-02-1542,

Copy Right Notice, and Proprietary Information Notice

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

Mr. Harv Hanneman
Nuclear Management Company
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

WEP-02-49
LTR-ESI-02-138

August 2, 2002

Westinghouse Sales Order 17320
NMC Order P001460 Line Item 004

NUCLEAR MANAGEMENT COMPANY
Point Beach Units 1 & 2
PBNP 1.4% MUR Power Uprate Support
Customer Affidavit For Withholding Proprietary Information (CAW-02-1542)

Dear Mr. Hanneman:

Please find enclosed the Westinghouse documentation to support the responses to the NRC Request for Additional Information (RAI) on the PBNP 1.4% Measurement Uncertainty Recapture (MUR) Power Uprate. Westinghouse scope for the Set 1 question 4 (Mechanical and Civil Discussion Points – 6 Questions).

This letter transmits the necessary information for PBNP's submittal of Westinghouse proprietary information related to the MUR Power Uprate Project. The following four documents are attached for your use in preparing the NRC submittals (Enclosure 1):

1. Information which should be included in your NRC transmittal letter.

2. Proprietary Information Notice to be attached to your NRC transmittal letter.

3. Copyright Notice to be attached to your NRC transmittal letter.

4. Westinghouse letter "Application for Withholding Proprietary Information from Public Disclosure" (CAW-02-1542) with Affidavit CAW-02-1542.

Please transmit the original of Item 4 to the NRC in your transmittal.

If you have any questions about this information, please contact Tim Kitchen on 412-374-4153 or myself.

Very Truly Yours,

WESTINGHOUSE ELECTRIC COMPANY

Lauren Brown for

Steve Swigart
Customer Projects Manager

enclosures

Cc:	L. Gunderson (No Enclosure)	Point Beach	1L

ENCLOSURE 1

Information For NRC Transmittal Letter

Proprietary Information Notice For NRC Transmittal Letter

Copyright Notice For NRC Transmittal Letter

Westinghouse Letter "Application For Withholding Proprietary Information From Public Disclosure (CAW-02-1542)," With Affidavit CAW-02-1542

Information For NRC Transmittal Letter

The following paragraphs should be included in your letter to the NRC:

Enclosed are:

1. "Point Beach Units 1 & 2 1.4% MUR Power Uprate Support NRC RAI Responses Set 1 Question 4" (Proprietary).

2. "Point Beach Units 1 & 2 1.4% MUR Power Uprate Support NRC RAI Responses Set 1 Question 4" (Non-Proprietary).

Also enclosed are a Westinghouse authorization letter, CAW-02-1542 accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 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 Section 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 Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-02-1542 and should be addressed to H. A. Sepp, Manager, Regulatory And Licensing Engineering, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Proprietary Information Notice For NRC Transmittal Letter

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 10CFR2.790 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 identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10CFR2.790(b)(1).

Copyright Notice For NRC Transmittal Letter

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Westinghouse

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

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Samuel J. Collins

Direct tel: (412) 374-5282
Direct fax: (412) 374-4011
e-mail: Sepp1ha@westinghouse.com

Our ref: CAW-02-1542

July 30, 2002

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Point Beach Units 1 & 2, 1.4% MUR Power Uprate Support, NRC RAI Responses
Set 1 Question 4

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1542 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.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1542 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry 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:

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 31st day
of July, 2002

Lorraine M. Piplica
Notary Public

LORRAINE M. PIPLICA
COMMONWEALTH OF PENNSYLVANIA
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 Westinghouse Electric Company, LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR 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 the 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, utilizes 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 constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of the following areas of potential competitive advantage:

(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, 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 diminishes 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 10 CFR 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.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Point Beach Units 1 & 2, 1.4% MUR Power Uprate Support, NRC RAI Responses Set 1 Question 4", July 2002, for Point Beach Units 1 & 2, being transmitted by the 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 the Nuclear Management Company for the Point Beach Nuclear Plants Units 1 & 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of uprating.

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

(a) Provide documentation of the methods for determining acceptable plant operation at uprate conditions.

(b) Provide specific analysis or evaluation results related to the parameters that are considered for the uprate project.

(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.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar uprating and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods.

Further the deponent sayeth not.