NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

NRC-03-047

10 CFR 50.90

April 30, 2003

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

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST 193, MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE FOR KEWAUNEE NUCLEAR POWER PLANT

References:

1) Letter NRC-03-004 from Thomas Coutu to Document Control Desk, "License Amendment Request 193, Measurement Uncertainty Recapture Power Uprate for Kewaunee Nuclear Power Plant," dated January 13, 2003 (TAC No. MB7225).

2) Letter to Mr. Thomas Coutu from John G. Lamb, "Kewaunee Nuclear Power Plant – Request for Additional Information Regarding Proposed Measurement Uncertainty Recapture Power Uprate (TAC NO. MB7225)," dated April 11, 2003.

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) submitted license amendment request (LAR) 193 (reference 1) for a measurement uncertainty recapture (MUR) power uprate of 1.4 percent. The MUR power uprate would change the operating license and the associated plant Technical Specifications (TS) for the Kewaunee Nuclear Power Plant (KNPP) to reflect an increase in the rated power from 1650 MWt to 1673 MWt.

On April 11, 2003, the Nuclear Regulatory Commission (NRC) issued requests for additional information (RAIs) regarding the proposed MUR power uprate (reference 2). This letter, with attachments, contains the NMC responses to the NRC formal RAIs. The following table details the attachments to this letter.

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Attachment	Content Description
1	Non-proprietary responses to the requests for additional information.
2	Excerpt (Section 8.4.2) from WCAP-16040-NP, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," Non-proprietary, February 2003.
3	Westinghouse Tables for Support of WCAP-15591, Revision 1 (Proprietary).
4	Westinghouse Tables for Support of WCAP-15591, Revision 1 (Non-proprietary).
5	Westinghouse authorization letter, CAW-03-1632, an accompanying affidavit, proprietary information notice, and copyright notice for attachment 3.
6	Figures supporting the I&C responses regarding the plant process computer system and UFMD installation.
7	Responses to Kewaunee CROSSFLOW RAIs (Proprietary).
8	Responses to Kewaunee CROSSFLOW RAIs (Non-proprietary).
9	Westinghouse authorization letter, CAW-03-1633, an accompanying affidavit, proprietary information notice, and copyright notice for attachment 7.

As attachments 3 and 7 contain information proprietary to Westinghouse Electric Company, they are supported by affidavits (attachments 5 and 9) signed by Westinghouse, the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the commission and address, 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 the supporting Westinghouse affidavits should reference the appropriate authorization letter and be addressed to H. A. Sepp, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

These responses to the RAIs do not change the Operating License or Technical Specifications for the KNPP nor does it change any of the proposed changes to the Operating License or Technical Specifications in reference 1. This response also does not change the no significant hazards determination or the environmental considerations originally submitted in reference 1. No new commitments are being made as a part of this response.

In accordance with 10 CFR 50.91, a copy of this letter, 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 April 30, 2003.

Thomas Coutu

Thomas Coutu
Site Vice-President, Kewaunee Plant

LMG

Attachments:
1. Non-Proprietary Responses to the Requests for Additional Information
2. Excerpt (Section 8.4.2) from WCAP-16040-NP, Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," Non-proprietary, February 2003
3. Westinghouse Tables for Support of WCAP-15591, Revision 1 (Proprietary)
4. Westinghouse Tables for Support of WCAP-15591, Revision 1 (Non-proprietary)
5. Westinghouse Authorization Letter, CAW-03-1632, Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 3
6. Figures Supporting the I&C Responses (Figure 1, Figure 2, and Drawing M-205)
7. Responses to Kewaunee CROSSFLOW RAIs (Proprietary)
8. Responses to Kewaunee CROSSFLOW RAIs (Non-proprietary)
9. Westinghouse Authorization Letter, CAW-03-1633, Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 7

cc- US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT 1

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Non-proprietary Responses to Requests for Additional Information

Questions from the Mechanical and Civil Engineering Branch

1. **On page 35 of Attachment 2 to the application, the licensee states that thermal, pressure, and flow rate "change factors" were developed during the evaluation of Nuclear Steam Supply System (NSSS) piping, other than the reactor coolant loop (RCL) and pressurizer surge line piping (e.g., chemical and volume control, residual heat removal (RHR), safety injection, internal containment spray, and component cooling water systems). The licensee also indicated that if the change factors were less than or equal to a five-percent increase, the increase was considered to be acceptable. If the change factors were greater than five percent, more detailed evaluations were performed. It is not clear how the five-percent criterion was determined as a cutoff percentage increase for evaluation of power uprate effects on piping.**

   Provide a summary describing how the calculations of the change factors are done with respect to the increase in temperature, pressure, flow rate, and thermal/pressure transients for the proposed power uprate. Also, provide a summary of the quantitative evaluation to confirm there are more than five-percent safety margins for systems that were evaluated using the change factors.

   NMC Response:

   Change factors were based on ratios of the power uprate and pre-power uprate system operating data for temperature, pressure, and flow rate. The change factor ratios are more fully described in attachment 2 to this letter that contains an excerpt from the Balance of Plant (BOP) report section 8.4.2 of WCAP-16040-NP, "Kewaunee Nuclear Power Plant NSSS and BOP Licensing Report." Section 8.4.2, "Description of Evaluation and Analysis," from WCAP-16040-NP provides a description of how the change factors were calculated. Fluid transient events were evaluated using separate analysis. For the systems listed in this question, no fluid transient events were identified, therefore, only the change factor analysis applied.

   Attachment 2, Section 8.4.2, describes the acceptability of using the change factor analysis. For change factors less than 1.05, no analysis is required since the current piping analysis is considered to have enough conservatism to cover a change factor of five percent or less. Again, these conservatisms are described in attachment 2. It is important to note that a change factor of 1.05 percent does not correlate to a five percent safety margin. Margin in the piping system calculations is not specifically calculated using the change factor approach.

The change factor methodology has been previously approved by the Nuclear Regulatory Commission (NRC) for use in power uprates at several other nuclear power stations including Turkey Point, Byron, Braidwood, Dresden, and Quad Cities. This change factor methodology is also applicable for the power uprate at the Kewaunee Nuclear Power Plant (KNPP) because, like the above listed power stations, (1) KNPP has utilized similar, simplified, industry standard methods such as manual calculations and linear elastic computer techniques in piping analysis, (2) KNPP has qualified piping and supports in accordance with USAS B31.1 Power Piping Code and other applicable codes, and (3) KNPP has utilized and installed appropriate materials in piping systems and supports.

2. **Section 5.5 of Attachment 3 to the application, the licensee indicated that RCL piping analyses were performed for an uprated power level of 1772 megawatts thermal (MWt) in compliance with United States of America Standard (USAS) B31.1, "Power Piping Code," 1967 edition, which is the code of record for RCL piping, and does not require a fatigue analysis. The acceptance criteria for the pressurizer surge line is based on the American Society of Mechanical Engineers' *Boiling and Pressure Vessel Code* (ASME Code), Section III, Subsection NB, 1986 edition, which is the code of record Kewaunee. The calculated stresses of RCL piping for the proposed power uprate condition are provided in Table 5.5.1-2 of Appendix 3 to the application.**

Provide the calculated maximum stresses and fatigue cumulative usage factors (CUFs) in compliance with the code of record for the pressurizer surge line piping for the proposed power uprate condition.

NMC Response:

As identified in reference 1, attachment 3, section 5.5.1.4, the current design basis results at 1650 MWt for the pressurizer surge line, including the effects of thermal stratification, are documented in WCAP-12841, "Structural Evaluation of the Kewaunee Pressurizer Surge Line, Considering the Effects of Thermal Stratification." WCAP-12841 was submitted to the NRC by letter on November 14, 1991 (reference 2) in response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." The associated NRC safety evaluation is documented by letter dated January 3, 1992 (TAC No. M72140) (reference 3). Westinghouse determined that the values in the WCAP-12841 are still applicable for the 7.4-percent uprate program (1772 MWt). The maximum stresses and usage factor for the pressurizer surge line from the WCAP are summarized below.

	Stress (ksi)	Allowable (ksi)
Equation 12 (*)	52.3	53.0
Equation 13 (*)	44.2	50.1
Usage Factor (*)	0.97	1.00

* As defined per Section NB3653.6 of the American Society of Mechanical Engineers' (ASME) Boiling and Pressure Vessel Code.

3. **In Table 5.8-1 of Attachment 4 to the application, the licensee compares the design parameter change in temperature, Delta T_{cold} (or Delta T_{hot}), between the pressurizer temperature and the cold leg (or hot leg) temperature. The higher the Delta T_{cold} is, the higher the thermal stress will be in the spray nozzle and the pressurizer upper shell. In Table 5.8-3 of Attachment 4 to the application, the licensee compares the original and revised stress intensity (SI) ranges for the proposed power uprate. The original design-basis SI range was calculated based on design-basis condition with a Delta T_{cold} of 125 °F in comparison to 132 °F for the proposed power uprate condition and 160 °F for the replacement steam generator condition.**

   Provide a summary of the evaluation, including stresses and CUFs, for the spray nozzle and the upper shell, which are exposed to higher Delta T_{cold} than they were originally designed for while operating at the proposed power uprate condition and in the replacement steam generator condition. Also, clarify how Delta T_{cold} is considered as 132 °F for the proposed power uprate condition and 160 °F for the replacement steam generator condition, while both the pressurizer temperature and the cold leg temperature are the same for both cases.

   NMC Response:

   The spray nozzle is a critical component. The stress levels in the spray nozzle envelop the stresses in the adjacent shell, as such, the shell stresses were not recalculated for the uprated operating conditions. Likewise Table 5.8-2 of reference 1, attachment 4 provided a summary of the cumulative usage factors (CUF) for the spray nozzle, where the values for the spray nozzle would envelop the values for the shell. The values on these tables represent a cold leg temperature to pressurizer spray nozzle delta T (ΔT) of 160° (F) for the Unit Loading/Unloading transient case. The 160° (F) ΔT was the temperature range specified for the design of the replacement steam generators (RSG) for this transient condition. This ΔT value resulted from conservative RSG design temperatures, combined with the temperature variations for the specified design transients. Actual operating temperatures would have resulted in a much smaller ΔT. When actual operating temperatures were established for the uprated power operating conditions, it was determined that the maximum ΔT for the spray nozzle would be 132° (F). Since the design ΔT for the RSG program was greater than that for the uprated power operating condition, and since the component qualification addressed the 160°(F) ΔT for the Unit Loading/Unloading transient, the stresses and CUF calculated for the RSG program bound the parameters that would result from operation at the uprated power conditions.

4. **In Section 5.8.1.6 of Attachment 4 to the application, the licensee provides an evaluation based on the ASME Code, 1965 edition with addenda through summer 1966. In Section 5.8.1.5, the licensee states that an elastic-plastic analysis was performed in accordance with Section NB-3228.3 of the ASME Code.**

   Identify the code of record and code edition used in the power uprate analysis.

NMC Response:

The 1971 edition of the ASME Code was used for the elastic-plastic analysis, since this was the first year this exact type of analysis was put into the code. The 1965 through Summer 1966 Addenda, version of the code did not include the provisions for elastic-plastic analysis, so the later code was adopted for the evaluation.

5. **On page 36 of Attachment 2 to the January 13, 2003, application, the licensee states that assessment of the balance-of-plant (BOP) piping and supports (including main steam, condensate and feedwater, auxiliary feedwater, and steam generator blowdown systems piping) were performed for a power uprate at 1772 MWt (which is about 7.4 percent above the current rated power of 1650 MWt). The licensee concluded that the piping and pipe supports remain in compliance with the USAS B31.1.**

    Provide a summary of the evaluation for the BOP piping and supports (including calculation of the "change factor"), the calculated maximum stresses, and CUFs at critical locations evaluated for each system's piping for the proposed power uprate conditions, the allowable ASME Code limits, and the ASME Code and its edition used in the evaluation. If different from the code of record, provide a justification.

    Describe how the change factors were calculated and how the factors were used to predict the stress values. Also, discuss how the stress, based on the change factors, is to be combined with those stresses due to fluid transient events for which separate analyses were performed.

    Provide more details of the technical basis regarding the statement that "the current piping analysis is considered to have enough conservatism to cover a change of five percent or less." Identify each conservatism in the current analysis to demonstrate they can accommodate a five-percent increase in the change factors.

NMC Response:

The BOP piping systems review concluded that all piping systems remain acceptable and will continue to satisfy existing design basis requirements under uprated conditions in accordance with the Code of Record, USAS B31.1, Power Piping Code, 1967. A detailed fatigue evaluation is not required for USAS B31.1. No piping or pipe support modifications are necessary as a result of the increased power level. The 7.4 percent evaluation bounds the 1.4 percent measurement uncertainty recapture (MUR) power uprate.

The BOP piping systems review was performed using change factor analysis as were the piping systems listed in question 1. When using the change factor analysis, maximum stresses are not recalculated or predicted at the uprated power. Change factors were based on ratios of the power uprate and pre-power uprate system operating data for temperature, pressure, and flow rate. The change factor ratios are more fully described in attachment 2, which contains an excerpt from WCAP-16040-NP, "Kewaunee Nuclear Power Plant NSSS and BOP Licensing Report." WCAP-16040-NP, Section 8.4.2, "Description of Evaluation and Analysis," provides a description of how the change factors were calculated. For change factors less than 1.05, no analysis is required since the current piping analysis is considered to have enough conservatism to cover a change factor of five percent or less. The justification for using a change factor of 1.05 is also described in attachment 2.

Fluid transient events were evaluated using separate analysis. For the systems with fluid transients, the fluid transient results are not combined with the change factor analysis. Of the systems in this question, a fluid transient event was identified for only the main steam system. For the main steam system, piping analysis was reperformed. For the remaining systems, however, no fluid transient events were identified and only the change factor analysis applied.

Summaries of the main steam (MS), condensate, and feedwater (FW) systems piping and supports evaluations for the 7.4 percent power uprate are included in attachment 2. The summaries can be found in sections 8.4.2.1.1, "Main Steam," 8.4.2.1.3, "Condensate," and 8.4.2.1.4, "Feedwater." Each report section contains a table with the calculated change factors and discussion regarding change factors higher than 1.05. The MS system stress levels and allowables are included directly below. This table was included since the MS system was the one system for which a transient event was evaluated. The auxiliary feedwater and the steam generator blowdown systems summaries are located in the following paragraphs. These summaries are provided here because the change factors associated with each are equal to or less than one and required no additional evaluation for uprated conditions.

Table 1 - Maximum Pipe Stress Levels and Allowables for Main Steam

Criteria	**Allowable stress (psi)**	**Max stress**
Criteria 5	1.2 Sh = 21000 psi	20665 psi
Criteria 6	1.8 Sh = 31500 psi	23837 psi
Criteria 5 (Upset): Pressure + Weight + SRSS (OBE Earthquake + Fluid transients due to Main Steam TSV event)		
Criteria 6 (Faulted): Pressure + Weight + SRSS (DBE Earthquake + Fluid transients due to MS turbine stop valve closure event)		

Auxiliary Feedwater (AFW) System: The 7.4 percent power uprate does not change the operating temperature, pressure, or flow rate of the AFW system. Therefore, the change factor would be 1.0 (i.e., the pre-uprate condition equals the power uprate condition) and the piping and supports were concluded to continue to be acceptable for the power uprate conditions. The 7.4 percent uprate evaluation bounds the 1.4 percent MUR power uprate.

Steam Generator Blowdown (SGBD) System: The 7.4 percent power uprate will change only the operating temperature and pressure of the SGBD system. At uprated conditions, the operating temperature and pressure would be 509°F and 797 psia, respectively. The existing SGBD piping analysis has considered higher bounding operating temperature and pressure (i.e., 561°F and 1115 psia) for the pipe stress qualification. For SGBD, the change factors for temperature and pressure would be less than 1.0. Therefore, the SGBD system was concluded to be acceptable for the 7.4 percent power uprate conditions. The 7.4 percent uprate evaluation bounds the 1.4 percent MUR power uprate.

Table 2 - Pipe Stress Margin Sample Requested during April 22 Conference Call

System	Design Basis (psi)	Baseline (psi)	Uprate (psi)	Allowable stress (psi)
Main Steam	15122	20509	20665	21000
Feedwater	12403	12403	12651	22500
Extraction at Inlet of FW Heater 11A/B	5050	5050	5060	15000
Extraction at Inlet of FW Heater 12A/B	5150	5150	5180	15000
Extraction at Inlet of FW Heater 15A/B	7700	7700	8000	15000

The results for extraction piping at the inlet to feedwater heater 15 A/B are presented (in addition to those for extraction piping to FWH 11A/B and 12A/B) because the stresses and changes due to uprate are the most limiting of the results shown in the WCAP-16040-NP table. No formal hand calculations or computer analyses have been identified, retrieved or located for the extraction piping. The assessment presented above is based on engineering judgment and construction practice for the installation of turbine plant piping systems. More specifically, the deadload stress for all three line entries for the extraction piping is considered to be 5000 psi or less. Normal industry "good practice" for reasonable spans for support of piping systems for deadload results in acceptable deflections and usually is consistent with stresses of about 1500 psi. To consider the deadload stress to be as high as 5000 psi is a conservative consideration in determining a design margin since we know the support scheme for extraction piping at KNPP is acceptable and meets the industry standard for deadload support spans. The balance of the stress contribution for extraction piping shown above in the first two columns is a representative hand calculation of longitudinal pressure stress (50 psi, 150 psi and 2700 psi, respectively) in accordance with the USAS B31.1 code. The increase in the third column for uprated stresses for all three rows of extraction piping shows a nominal increase in stress (10 psi, 30 psi and 300 psi respectively), due entirely to a nominal increase in operating pressure which results in a nominal increase in longitudinal pressure stress.

The fourth column for all five rows of data presents the code allowable for carbon steel piping based on the allowable stress tables in USAS B31.1.

The critical stress for main steam piping is due to sustained plus occasional stresses. The critical stress for feedwater piping is due to thermal expansion. The critical stresses for extraction piping are all due to sustained stresses.

Note that for the main steam piping, the existing design basis calculation does not provide specific stress results for the turbine stop valve closure event. That is, no detailed analysis had been previously performed for this event for KNPP. This is why a separate entry, the baseline stress case, is presented (and is different from the current design basis stress).

Questions from Electrical and Instrumentation Controls

Clarification to Table V-1, Electrical Equipment Information

Table V-1, "Electrical Equipment Information," was provided for NRC review in reference 1, attachment 2, page 49. During conference calls with the NRC staff on April 9 and April 21, 2003, the Nuclear Management Company (NMC) provided further clarification of the values in the sixth column titled, "Anticipated Power Uprate (7.4%)." The clarification was that although most of the values were anticipated values for the 7.4 percent power uprate, some values were determined using worst case or bounding conditions. In particular, clarification was provided for the Main Auxiliary Transformer (MAT) Iso-phase Bus. The MAT Iso-phase Bus anticipated power uprate amperage for Table V-1 was calculated using the maximum MAT rating of 44.8 MVA (the existing design limit from column 5) assuming the maximum current through the Tap occurs with the MAT at full capacity and the generator voltage at its low limit. Maximum current is calculated as 1,360 amps. However, if using the anticipated MAT operating value at the power uprate (from Column 6, i.e., 32.3 MVA) in place of the maximum MAT rating, the MAT iso-phase bus anticipated value at power uprate becomes 982 Amps.

General question

1. **Please explain how the plant process computer (PPC) reactor thermal output (RTO) calculation is used in the operation of the plant.**

   NMC Response:

   The Plant Process Computer System (PPCS) calculation of reactor thermal output (RTO) is used for monitoring reactor power such that operators can control power less than the licensed limit. It is important to note that the PPCS RTO calculation is used for indication only and does not perform any safety related functions and is not used to directly control any plant systems. The PPCS RTO calculation is used for the daily nuclear power range calibration (Table TS 4.1-1).

Questions pertaining to Attachment 2 of the application

1. **Section I.1.C (page 2) states, in part, "...the RTO computer program will be modified to receive the Crossflow UFMD [ultrasonic flow meter device] generated individual venturi flow and ..."**

   Explain the phrase "UFMD generated individual venturi flow." Also, discuss the RTO computer program change, in detail, to include changes in the calculation algorithm and communication interface (e.g, will the change or interface add additional uncertainties? Why or why not?).

NMC Response:

The phrase, "UFMD generated individual venturi flow and temperature correction factors for use in the RTO calculation program," is found in attachment 2, page 2, section I.1.C. This statement means that the RTO computer program will receive the ultrasonic flow measurement device (UFMD) generated correction factors for the individual feedwater flow channels on the A and the B feedwater loop. Likewise, the RTO computer program will receive individual generated temperature correction factors for both the A and B feedwater loops.

The PPCS provides the UFMD cabinet feedwater flow signals from both feedwater channels on each feedwater loop and a feedwater temperature signal on each feedwater loop. The UFMD cabinet compares the loop flows from the PPCS with the loop flow measured by the ultrasonic flow meter (UFM) and derives a specific correction factor for each flow channel. The UFMD cabinet also compares the loop temperature from the PPCS to the temperature measured by the loop ultrasonic temperature measurement (UTM) instrument and derives a loop-specific temperature correction factor. These flow and temperature correction factors are returned to the PPCS and used to develop the corrected feedwater flow and corrected feedwater temperature inputs to the RTO program. The PPCS and UFMD inputs are also described in detail in the response to question 3 for reference 1, attachment 7 (the next set of I&C related questions).

The RTO calculation in the PPCS RTO program itself will not change. The only change is that the PPCS feedwater flow and temperature values currently corrected with manual correction factors will be replaced with PPCS feedwater flow and temperature values corrected with the new UFMD correction factors.

2. **Section I.1.C (page 2) states, in part, "...installation of [Crossflow] ...meets the requirements of CENP-397-P-A (reference I.1, section 1.4.2)." There do not appear to be requirements in this section. What requirements is the attachment referring to?**

   NMC Response:

   This statement was intended to mean that the installation of the temporary UFMD on the full flow feedwater bypass line can be used in conjunction with the A loop UFMD to calibrate the B feedwater loop UFMD, the later of which did not meet the vendor's requirements for fully developed flow. Both the full flow bypass line and loop A met the requirements from the vendor for this type of calibration. The vendor's requirements are that the calibration installation be installed in a line that is considered to have fully developed flow as defined in CENPD-397-P-A, Section 8.1.1.

3. **The licensee proposes the use of ultrasonic temperature measurements. As stated in Section I.1, these are "...not described in CENP-397-P-A ["Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology]." Figure 3 of Attachment 7 indicates use of the UTM [ultrasonic temperature measurement] for density correction and enthalpy of the feedwater. Furthermore, Table I.1 (page 9) suggests an additional 0.2% reactor thermal power to be gained by using the UTMs.**

Provide information on the use of the UTMs in sufficient detail for the NRC staff to evaluate its use. The following are some items to include:

- **The type of sensor and the theory of operation**
- **Is it an analog or digital sensor?**
- **A diagram of the sensor (preferably from the vendor drawings)**
- **Make, model number, etc.**
- **How the sensitivity and uncertainty values were determined (i.e., where do the numbers come from)**
- **How the UTM interfaces with the PPC and have those interfaces been considered in the uncertainty determination**
- **How the UTMs will be calibrated and what is the suggested calibration periodicity**

NMC Response:

Refer to the response found in question 3 of attachment 7 to this letter.

4. **CENPD-397-P-A identifies and details several diagnostic features associated with the ultrasonic flow meter (UFM) providing the operators with information regarding its availability. Section I.H (page 8) states, in part, "If the UTMs...become unavailable." However, there is no discussion of the UTM in CENPD-397-P.**

What indications will be available to plant operations staff letting them know that the UTM or UTMs are unavailable? Provide a detailed enough response to allow the NRC staff to understand how long the plant would be operating at the additional margin afforded by use of the UTMs before their failure is discovered by plant operations staff.

NMC Response:

The only discussion pertaining to UTMs in the topical report is a statement that improving the accuracy of the feedwater temperature instrumentation can improve the density term in feedwater flow determination. This can lead to more accurate density measurement and lower total feedwater flow measurement uncertainty. The NMC decided to implement the use of higher accuracy feedwater temperature measurement instrumentation (e.g., the UTMs).

The indication of an unavailable UTM correction factor is the same as for an unavailable UFM. The PPCS indications and alarms are the same as described in the response to question 4 in the next group of I&C questions pertaining to reference 1, attachment 7. As soon as the computer determines that a correction factor is questionable, the effected correction factor is identified by a "Q" besides the value, the UFMD OPERATING LIMIT display is automatically reduced to the level associated with the appropriate instrument, and the audible and visual alarm TLA (trouble light alarm) 28, "POWER GREATER THAN UFMD LIMIT," is actuated. The TLA 28 is located in the control room, immediately notifying operators of this condition. Refer to question 4 in the next group of questions for additional detail.

5. **Section 8.1.3 of CENPD-397-P-A discusses the transducer installation.**

Discuss any expected differences in the mounted-to-a-support-frame (M/TSF) area temperature from the time it is measured at installation to the time Kewaunee will be operating.

NMC Response:

Refer to response found in Attachment 7, Question 3.

6. **Discuss, in further detail than provided on page 3 (D.1), the maintenance items the licensee foresees for the UFMD. For example, in CENPD-397-P-A, Appendix B, "Response to NRC Request for Additional Information Supporting Topical Report CENPD-397-P Review Activities," the RAI-13 response discusses an "internal time delay check [which] is confirmed monthly in the field...."**

The licensee's response can be in the form of a tabularized or bulleted list which can provide the NRC staff with assurance that preventative maintenance has been appropriately identified.

NMC Response:

The "internal time delay check" mentioned in this question is now part of the self-test software of the signal conditioning unit (SCU) and will be performed automatically by the UFMD during normal operation. Therefore, it is no longer necessary to schedule a "Monthly Internal Time Delay Check" as described in CENPD-397-P-A.

Being a completely digital system, the UFM only requires periodic checks of functionality using the built-in software as recommended by the vendor. The vendor will perform any required calibration. This is consistent with the site testing being performed by other utilities already using the Crossflow system.

The maintenance items anticipated for the Crossflow UFMD are summarized below. This maintenance plan has been reviewed by the UFMD vendor and the vendor agrees it meets the intent of the Topical Report and the CROSSFLOW Users Manuals.

<u>Ultrasonic Flow Measurement</u>

- Reboot the SPU (Signal Processing Unit) every two months.
- Perform SCU (Signal Conditioning Unit) self test monthly. This SCU monthly self test will be performed automatically with the UFM to PPCS interface software installed with the KNPP system.
- Perform the RSSI (Reflected Signal Strength Indication) scan after cold shutdown, startup, after a feedwater temperature change of >100 °F, and after a year of continuous operation. KNPP plans to perform this scan on a yearly basis and as required by the above statements.
- Recalibrate the SCU every refueling outage (approximately 18 months) by returning to the vendor.

Ultrasonic Temperature Measurement:

- Reboot the SCP (Signal Conditioning/Processing Unit) every two months.
- Recalibrate the SCP every refueling outage by returning it to the vendor for calibration with the SCU.
- Perform hard disk maintenance yearly.

The KNPP has committed to update or provide new documents (i.e., procedures) for changes associated with the installation of the Crossflow UFMD in the original 1.4 percent MUR power uprate submittal (reference 1). This includes the incorporation of the maintenance requirements listed above. No new commitments are made with this response.

7. **In Section F.v (page 7), the licensee addressed Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Section F.v for the UFMDs only.**

Address Item I.1.F.v receiving and addressing manufacturer deficiency reports for the remaining "instruments that affect the power calorimetric."

NMC Response:

Any deficiency reports sent to KNPP for instruments used at the plant are screened and addressed through the Operating Experience Assessment (OEA) process. This would include the instrumentation used in the calorimetric power determination.

Questions pertaining to Attachment 7 of the application (WCAP-15591, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Kewaunee Nuclear Plant (Power Uprate to 1757 MWt - NSSS Power with Feedwater Venturis, or 1780 MWt - NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators")

1. **Explain the correlation between calorimetric sensitivities and associated measurement uncertainties given in Tables 10 - 15 and provide the methodology used to determine the individual sensitivities and uncertainties for values given in these tables.(e.g., where do the values come from?)**

   NMC Response

   The uncertainty is determined for an instrument channel (see reference 1, attachment 7, Table 10) consistent with ISA S67.04. The sensitivities (see reference 1, attachment 7, Table 11) are based on the slope of a steam table curve (NBSNRC Steam Tables) at the design operating conditions. The instrument channel uncertainty is multiplied by the corresponding sensitivity to determine its impact on the power measurement uncertainty. The power measurement uncertainty components (see reference 1, attachment 7, Table 12) are statistically combined to determine the total power measurement uncertainty. This methodology is consistent with NUREG/CR-3659 dated February 1985.

   - **On page 3, which provides equations 1, 2, and 3, why are the variables of allowance for conversion accuracy of an analog-to-digital signal for PPC use not present?**

     NMC Response:

     Equation 2 on page 3 of reference 1, attachment 7, is the general form of the uncertainty for a plant computer measurement. Uncertainties with the subscript (comp) are for the A/D portion of the computer.

   - **Table 1 (page 8) provides control and indication uncertainties. The NRC staff calculated a value of 5.25 percent for channel statistical allowance (without BIAS value) and thus calculated a different value for pressurizer pressure electronics uncertainty. Provide an explanation of the difference.**

     NMC Response:

     See Table 1 in attachment 3 of this letter, "WCAP-15591, Rev 1 (in support of Table 1)."

- **Table 3a (page 19) provides flow uncertainties. Why are values not present for (1) the reference signal uncertainty for a closed-loop automatic control system, (2) measurement and test equipment used to calibrate the controller rack module(s) that perform the comparison between the controlled parameter and the reference signal, and (3) allowance of the controller rack module(s) that perform the comparison and calculates the difference between the controlled parameter and the reference signal?**

  NMC Response:

  The reference signal uncertainty for a closed-loop automatic control system (REF), the measurement and test equipment used to calibrate the controller rack module(s) (CMTE), and the allowance of the controller rack module(s) (CA) are uncertainties associated with an automatic control system instrument loop. Page 19 of WCAP-15591, Rev 1 (reference 1, attachment 7) represents a once-per-fuel-cycle calorimetric reactor coolant system (RCS) flow measurement that does not use automatic control system instrument loops.

- **On page 19, the NRC staff calculates a different value than 3.6 in the table. The 3.6-percent channel statistical allowance multiplied by the span of 800 does not equal 14.2. It appears that the number 3.6 is divided by the square root of 4, and multiplied by the span, but that value is equal to 14.4. Provide an explanation of the difference.**

  NMC Response:

  See Table 3a in attachment 3 of this letter, "WCAP-15591, Rev 1 (in support of Table 3a)."

- **What equation is being used on page 33 and why are some uncertainty values being subtracted?**

  NMC Response:

  The equation on page 33 of WCAP-15591, revision 1 (reference 1, attachment 7) is the square-root-sum-of-the-squares combination of the power measurement uncertainty components of Table 9 (page 37 of reference 1, attachment 7) and notes two sets of interactive effects. The first set is feedwater temperature that provides an uncertainty input into Feedwater Flow Thermal Expansion Coefficient, Feedwater Flow Density and Feedwater Enthalpy. An error in feedwater temperature is common to all three parameters and can result in an indicated lower than actual effect for two terms (that are added) and an indicated higher than actual effect for the other term. The second set is feedwater pressure that provides an uncertainty input into Feedwater Flow Density and Feedwater Enthalpy. An error in feedwater pressure is common to both parameters and can result in an indicated lower than actual effect in one parameter and an indicated higher than actual effect in the other parameter, resulting in a subtraction effect.

- **Page 40 presents uncertainties of steam generator blowdown. Why are some values not present for the turbine flow meters (e.g., sensor pressure effects and primary element accuracy)?**

  NMC Response:

  The accuracies on page 40 of reference 1, attachment 7, represent the total errors for the turbine flow meters as reported by the equipment manufacturer. The total error is reported as the CSA. There are no specific values for SPE and PEA for the turbine flow meters.

2. **Have uncertainties associated with the UFM and UTM communication protocol to the PPC been considered? If yes, are they included in the calculation? If yes, identify which part.**

   NMC Response:

   The UFM and UTM Algorithm and Communication Layer (ACL) uncertainties to the plant computer have been considered and are included in the 0.5 percent of flow accuracy and the 1.1 degrees-F temperature accuracy. The ACL is effectively a calculator that manipulates existing digital signals taken from the plant computer. The interface between the UFMD system and the PPCS uses a transmission control protocol/internet protocol (TCP/IP). The TCP/IP guarantees accurate delivery of the information. There is no analog-to-digital conversion.

3. **Provide a diagram similar to figures 2 and 3, but in more detail to identify calculations performed in the PPC or elsewhere, inputs to the PPC, and the communications links that exist. Also, provide the communications protocols for all inputs (e.g., analog-4-20 ma, EIA-232, 422). An engineering-grade drawing is preferable.**

   NMC Response:

   Figure 1 of attachment 6 to this letter shows the PPCS and UFMD electronics cabinet inputs and outputs and communication links that exist for the measurement of the feedwater flow and temperature for the A feedwater loop. The B feedwater loop communication inputs and outputs are identical. The interface between PPCS and the UFMD system is purely digital and contains essentially no error. The communication links are described in detail in the following paragraphs.

   The PPCS interface data link provides data between the PPCS computer and the UFMD electronics cabinet. This data link provides the required plant data to the UFMD cabinet to allow the UFMD electronics cabinet to generate correction factors for feedwater flow and feedwater temperature for each steam generator. There are two individual feedwater flow correction factors calculated for each loop because there are two venturi flow channels on each feedwater loop. There is one temperature correction factor calculated for each feedwater loop. In addition, the interface data link returns the feedwater flow and temperature correction factors and UFMD status (quality) signals to the PPCS.

The UFMD electronics cabinet software, CROSSFLOW for flow measurement and CORRTEMP for temperature measurement, collects data from the UFM and UTM sensors on the feedwater line to each steam generator to derive a total feedwater flow and feedwater temperature signal for each feedwater loop. This UFMD data is then compared to the feedwater flow and temperature signals provided from the PPCS to develop instantaneous and average correction factors for the feedwater flow instrument channels (UFM flow/PPCS flow) and feedwater temperature channel (UTM temp/PPCS temp) on each loop. The quality of the correction factor data is determined and saved along with the corresponding correction factor to a storage buffer. A moving average of the most recent correction factors stored in the storage buffer and an average of a greater sample is calculated for each of the correction factors. Plant specific database constants are inputs to the CROSSFLOW and CORRTEMP software to ensure the correction factors are maintained within the required uncertainty, limits, time constants, and plant values necessary to support the flow measurement uncertainty required by a MUR uprate.

Individual correction factors for flow (one for each channel, two per loop) will be provided from the UFMD electronics cabinet to the PPCS for use with the loop A PPCS feedwater flow values and the loop B PPCS feedwater flow values. The two corrected feedwater flows from each loop will be averaged developing the average corrected feedwater flow by loop for use in the RTO PPCS program. Individual correction factors will be provided for loop A PPCS main feedwater temperature and loop B PPCS main feedwater temperature. The Figure 1 schematic of attachment 6 shows the above communication links for feedwater flow and temperature for loop A. The interface data link communication and calculations performed in the PPCS are the same for the B loop.

The only calculations that exist are those to develop the correction factor in the UFMD software and the combination of the correction factors with the PPCS flows and temperatures in the PPCS. The actual calculations performed in the RTO program are those described in reference 1, attachment 7 (WCAP-15591). In attachment 7, figure 3, the boxes for UTM and UFM indicate the corrected feedwater flow and temperature developed through the use of UFMD correction factor and the PPCS data points for feedwater flow and temperature.

The following two lists summarize the inputs and outputs to the UFMD electronics cabinet and PPCS.

UFMD Cabinet Inputs

- A loop PPCS main feedwater flow values and signal qualities derived from two venturi flow channels (from the PPCS).
- B loop PPCS main feedwater flow values and signal qualities derived from two venturi flow channels (from the PPCS).
- A Main Feedwater temperature and signal quality (from the PPCS).
- B Main Feedwater temperature and signal quality (from the PPCS).
- Main Feedwater common header pressure (density compensation for the UTM and UFM sensors) and signal quality (from the PPCS).
- Total A loop feedwater flow from the A loop UFM.
- Total B loop feedwater flow from the B loop UFM.
- Total A loop feedwater temperature from the A loop UTM.
- Total B loop feedwater temperature from the B loop UTM.

UFMD Cabinet Correction Factors Provided to the PPCS

- A loop feedwater flow correction factors (one for each feedwater flow channel) and signal quality information.
- B loop feedwater flow correction factors (one for each feedwater flow channel) and signal quality information.
- A loop feedwater temperature correction factor and signal quality information.
- B loop feedwater temperature correction factor and signal quality information.

4. How will the PPC implement the different suite of inputs (UFM/UTM, UFM/resistence temperature device (RTD), venturi/RTD) consistent with the operational conditions of the UFMs, UTMs and RTDs? Provide enough detail to aid the NRC staff in understanding how the operational modes are selected.

NMC Response:

The UFMD must be in service and providing good quality correction factors to the PPCS RTO program prior to increasing power above 1650 MWt. Operation at 1673 MWt requires both the feedwater flow and feedwater temperature correction factors to be in service. If either of the feedwater temperature correction factors is out of service, reactor power is administratively limited to 1670 MWt. If any of the feedwater flow correction factors are out of service, reactor power is administratively limited to 1650 MWt. The PPCS display screen, "UFMD/UTM CORRECTION FACTORS" (PPCS screen 60), automatically displays the UFMD OPERATING LIMIT based on the available correction factors. PPCS screen 60 also displays the status of the feedwater flow and temperature correction factors, the RTO OPERATING LIMIT, and the CURRENT REACTOR OUTPUT (a 15 minute sliding average from the RTO program). The RTO OPERATING LIMIT is the allowable value for limiting reactor thermal power. Figure 2 in attachment 6 of this letter provides the currently planned layout of PPCS screen 60.

Note: that this screen structure provided in Figure 2 may change as the KNPP operators obtain experience with the system.

In the event of a UFMD failure (i.e., inability to generate any one of the feedwater flow or feedwater temperature correction factors) the PPCS system is to use the last good correction factors prior to the UFMD failure. Therefore, the PPCS RTO program will use the corrected feedwater flows and temperature that were based on the last good correction factors. The last good correction factors provided by the UFMD will be used until the UFMD is returned to service and providing updated correction factors.

When either a feedwater flow or temperature correction factor from the AMAG becomes questionable (not updating or connectivity with the AMAG system is lost), a "Q" (i.e., questionable quality factor) appears next to the effected correction factor value on the PPCS screen 60. When the "Q" appears on PPCS screen 60, the UFMD OPERATING LIMIT display will automatically be reduced appropriately for the questionable correction factor. That is, the UFMD OPERATING LIMIT on PPCS screen 60 will be reduced to 1670 MWt if either of the feedwater temperature correction factors have a "Q" next to them and to 1650 MWt if any of the feedwater flow correction factors have a "Q" next to them.

If RTO OPERATING LIMIT is above the allowable UFMD OPERATING LIMIT, both displayed on PPCS screen 60, the computer TLA 28, "POWER GREATER THAN UFMD LIMIT," will alarm. The RTO OPERATING LIMIT will continue to show the allowable power until the next power range nuclear instrumentation (NI) surveillance (Table TS 4.1-1). If TLA 28 alarms during a load increase greater than 1650 MWt, the load increase must be stopped and load held at the existing value until the correction factor quality can be restored to good quality and is updating. An additional computer alarm will activate anytime the CURRENT REACTOR POWER (RTO program) exceeds the RTO OPERATING LIMIT.

If the failure of the UFMD is not corrected prior to the next scheduled power range NI daily surveillance, reactor power must be reduced as appropriate for the type (i.e., UTM or UFM) of failure. The last good correction factors from the UFMD will continue to be used by the PPCS RTO program during this time. Procedural guidance to the operators will be to manually apply the UFMD OPERATING LIMIT from PPCS screen 60. When the UFMD OPERATING LIMIT is applied, the RTO OPERATING LIMIT will automatically be reset to the UFMD OPERATING LIMIT value displayed on PPCS screen 60. Operation at this power level will continue until the UFMD correction factor is restored.

Once a correction factor is restored to good quality, the UFMD OPERATING LIMIT will automatically be updated to the maximum allowable power consistent with the status of all correction factors. Increasing the RTO OPERATING LIMIT to the UFMD OPERATING LIMIT requires the operator to manually apply the UFMD OPERATING LIMIT from PPCS screen 60.

5. **A number of calculations are present which support reactor coolant system (RCS) flow measurement.**

   Describe how use of these calculations will support the 1.4-percent power uprate (i.e., identify which calculations, pages, and tables (of Attachment 7) are being used to support the 1.4-percent power uprate).

Prior to the MUR power uprate project, the KNPP performs the RCS flow measurement using the venturi on the full flow feedwater bypass line. However, the installation of the Crossflow UFMD provides KNPP with an alternate means of measuring RCS flow. Reactor engineering at KNPP plans on implementing the use of the UFMD corrected feedwater flows for RCS flow measurement following the MUR power uprate. Implementation includes updating reactor engineering procedures to include the use of the UFMD. Note that the venturi on the feedwater bypass loop can still be used to measure RCS flow as long as nuclear steam supply system (NSSS) power is 1757 MWt or less. The MUR power uprate NSSS power is 1681 MWt. Therefore, either venturi flows or UFMD corrected flows can be used for calculation of RCS flow measurement.

6. **Provide the in-situ calibration procedure in sufficient detail for the NRC staff to understand the plant specific configuration and process. The following are some items/considerations to include:**

- **A piping and instrumentation diagram showing the location of proposed UFM and UTM sensors and stand-alone UFM with detail to include the feedwater bypass and flow paths (an engineering drawing (with ISA type symbols) may be the best way to accomplish this). This drawing can then be used in the discussion.**

  NMC Response:

  The location of the UFMDs on the A and B feedwater loops are upstream of the main feedwater regulating valves as shown on drawing M-205, "Flow Diagram Feedwater System," included in attachment 6 to this letter. The upper left hand quadrant of the drawing shows the feedwater flow paths to the A and B steam generators as well as the 24 inch, full flow bypass line (drawing coordinates A to C, 5 to 8). It is important to note that the temporary, stand-alone UFMD that was installed for calibration purposes on the permanently installed 24 inch, full flow bypass line is not shown on this drawing since it was a temporary installation and not a permanent modification to the plant. The temporary installation was for a one-time calibration of the B feedwater loop. That calibration is explained in the next bulleted item below. The temporary, stand-alone UFMD has already been removed.

- **A discussion of the calibration procedure to be performed and which UFMs need to be calibrated using the stand-alone UFM. Are there one or two calibration UFMs?**

  NMC Response:

  Following the initial installation, there will be no in-situ calibration at the site and there will be no calibration UFMDs. The vendor determined that the A feedwater line and the 24 inch, permanently installed bypass line met the acceptable piping configuration requirements for fully developed flow. However, the B feedwater line did not meet these requirements and required in-situ calibration. Therefore, development of a correction factor or multiplier was necessary for the B feedwater line. Note that neither the A feedwater line nor the full flow bypass line UFMDs required calibration when installed at the site because these lines met the requirements for fully developed flow as defined by CENPD-397-P-A, Section 8.1.1.

The temporary, stand-alone UFMD was installed on the 24 inch, full flow feedwater bypass line immediately upstream of the installed venturi flow nozzle FE27182 (see M-205 in attachment 6 for location of flow nozzle) to perform the in-situ calibration of the B feedwater line UFMD. Using the temporary, stand-alone UFMD on the full flow bypass line, total feedwater flow could be accurately determined. Flow through the A feedwater line could also be accurately determined. The flow through the B line was then determined by subtracting the flow determined for the A feedwater line from the total feedwater flow determined for the full flow bypass line. From this, a correction factor was developed. This correction factor will be applied to the B feedwater loop flow signal to provide an accurate B line flow. This in-situ calibration was a one time test to establish the B line correction factor and is not required again unless there is a physical change in the feedwater piping configuration.

Calibration of the UFMD signal conditioning unit is discussed in the next bulleted item below.

- **A discussion how often the UFMs are to be calibrated.**

  NMC Response:

  As stated above in the second bulleted item, neither the A nor B feedwater line UFMDs will continue to require calibration using the temporary, stand-alone UFMD. The temporary, stand-alone UFMD was used only to develop a correction factor for the B loop because the B loop did not meet the requirements defined in the topical report for fully developed flow.

  The A and B feedwater line UFMD Signal Conditioning Units (SCUs) will be periodically calibrated by the vendor. The UFMD system is an electronic hardware system where all functions are controlled and calculated by a computer processing unit, with no user adjustable components. The vendor will perform the periodic calibration of the SCUs. The site will return the processor unit to the vendor each refueling for calibration. The signal conditioning units require calibration every two years as stated in the topical report.

- **A determination to verify fully developed flow in the stand-alone UFM and UFMs that do not require stand-alone calibration.**

NMC Response:

The UFMD installed on the A loop and the temporary UMFD installed on the permanent full flow feedwater bypass line both meet the vendor's requirements for fully developed flow (acceptable piping configuration) as defined by CENPD-397-P-A, Section 8.1.1. The vendor determined this during an on-site visit. However, the B feedwater line piping did not meet the vendor's requirements for fully developed flow. Flow on the B feedwater line required correction as described above in the second bulleted item using the UFMDs installed on the A loop and on the permanent full flow feedwater bypass line.

- **A discussion of how the existing feedwater venturis will be calibrated to allow the operation of the reactor at power given in Table I.1.**

NMC Response:

The existing feedwater venturis and the associated loop instruments, including their calibration, will be unaffected by the UFMD and the 1.4 percent MUR power uprate. The feedwater flow channels for each venturi will continue to be calibrated as they have always been calibrated using the appropriate plant procedures and processes. The power measurement uncertainty for venturis remains the same as it did prior to the MUR power uprate (i.e., two percent).

7. **For installation of the calibration UFM, how will fully developed flow be assured in the feedwater bypass line given the use of a flow straightener, venturi nozzle, and diffuser?**

NMC Response:

Fully developed flow for the installation of the UFM on the full flow bypass line was confirmed based on an existing laboratory hydraulic model comparable to the Kewaunee piping configuration. The results of the laboratory test were confirmed through in-plant testing by taking flow measurements in both the vertical and horizontal planes. Since the difference between the two readings was well within the statistical uncertainty of the meter, it was concluded that the flow was indeed fully developed and met the requirements of the topical report, CENPD-397-P-A, Sections 5.6.1 and 8.1.1.

It should be noted that the diffuser section, which is upstream of the flow straightener, is actually a reducer with a step change in pipe wall of only 0.25 inches. With a flow straightener and 15.5 pipe diameters downstream of the diameter change, any effect due to the reduction in pipe would be easily dissipated. Furthermore, the venturi nozzle is located downstream of the meter, so any flow disturbance introduced by the venturi does not affect the meter.

8. **In CENPD-397-P-A, Appendix B, "Response to NRC Request for Additional Information Supporting Topical Report CENPD-397-P Review Activities," the RAI-9 response discusses the effect of corrosion products on UFM measurement. The ABB Combustion Engineering Nuclear Power response provides recommendations to address pipe-wall monitoring.**

Discuss the evaluation performed or planned to address this RAI response and CENPD-397-P-A, Section 5.4, Inside Pipe Diameter.

NMC Response:

Throughout the industry, there is little evidence of pipe wall thinning in the straight runs of feedwater piping. Wall thinning typically occurs downstream of elbows, valves, and fittings, where there is maximum turbulence. Since the UFMD is installed in straight runs of piping, erosion is kept to a minimum.

To provide further protection, NMC has reviewed their flow accelerated corrosion program and has determined that feedwater pipe thinning in the area of the UFMD installations has not been a problem. However, if the flow accelerated corrosion program indicates that pipe wall thinning occurs at a point of higher turbulence in the feedwater system, NMC would consult with AMAG on the potential impact on the UFMD meters.

Deposition of corrosion products on the feedwater pipe internal surfaces would bias the flow reading in the conservative direction. Any change (increase) in wall thickness due to the deposition of corrosion products would have a conservative affect on the UFMD measurement of feedwater flow. KNPP feedwater chemistry is maintained to prevent buildup of corrosion products and buildup of corrosion products is not a likely problem.

9. **How are the velocity profile curves for each loop determined at Kewaunee?**

NMC Response:

The velocity profiles are based on the assumption of smooth wall pipe and fully developed flow. The A feedwater loop meets these requirements as defined in CENPD-397-P-A, Section 8.1.1, while laboratory and in-plant testing confirmed that the full flow bypass line also met the requirements for fully developed flow. However the B loop did not meet these requirements because of upstream flow disturbances. Therefore, the B loop was calibrated in-place by measuring both the full flow bypass (total feedwater flow) and A loop flow. The difference between the full flow bypass and A loop was then used to calibrate the B loop UFMD. This approach provides the most accurate calibration, since it was performed under actual full power operating conditions.

10. **Since the Crossflow is a clamp on flow meter, describe the interactions (environmental, physical, fluid communication) that will affect existing feedwater instrumentation due to the installation of the Crossflow system and any interactions that existing flow instrumentation will have on the Crossflow system.**

NMC Response:

There will be no interactions from the Crossflow UFMDs that will affect the existing feedwater instrumentation or vice versa. This is based on the fact that the UFMD instrumentation is physically located away from the feedwater venturi instrumentation.

11. In Table 10 (page 40) for UTM feedwater temperature, why is there no uncertainty for allowance for conversion accuracy of an analog-to-digital signal for PPC use? Is this already considered in the uncertainty provided to the licensee?

NMC Response:

There is no analog-to-digital conversion for UTM feedwater temperature measurement. The UTM provides a digital signal to the plant computer for the feedwater temperature correction factor. The total UTM feedwater temperature measurement uncertainty is included in the reported value.

Materials And Chemical Engineering Questions:

Tube Repair Limits (Regulatory Guide 1.121 Analysis)

1. **In Section 5.7.10 of Attachment 3 to the application, the licensee indicates that an analysis is being performed to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. Calculations have also been performed to establish the structural limit for tube straight leg (free span) flaws over an unlimited axial extent and for degradation over limited axial extent at the tube support plate and antivibration bar intersections. As part of Kewaunee License Amendment No. 158, dated May 25, 2001, related to replacement steam generators, the licensee indicated that WCAP-15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Steam Generators," showed that the existing through-wall repair limits remained conservative for the replacement steam generator tubes.**

   NMC Response:

   Tube structural limits are defined for the Kewaunee replacement steam generators in WCAP-15325, "Regulatory Guide 1.121 Analysis for the Kewaunee Steam Generators." A revised analysis was performed to document applicable tube structural limits for the uprated conditions. The analysis results show that, although the primary-to-secondary pressure gradients are increased for the uprated conditions, the changes were not large enough to result in a change to the structural limits. As a result, the tube structural limits of WCAP-15325, remain applicable to the uprated conditions.

2. **In many sections of the application, it states that the conditions are bounded by the analysis developed for that Steam Generator Replacement Project.**

   Provide a copy of the Steam Generator Replacement Project Analysis and 10 CFR 50.59 safety evaluation for the applicable sections that are referenced in the MUR power uprate submittal.

   NMC Response:

   During a conference call between KNPP and NRC staff on April 22, 2003, it was determined that the applicable sections of the Replacement Steam Generator (RSG) Licensing Report and KNPP RSG 10 CFR 50.59 safety evaluation did not require submittal to the NRC to complete the MUR power uprate review. The KNPP RSG 10 CFR 50.59 safety evaluation references the SGR project report. Both documents are retrievable at the KNPP site and are available to the NRC for audit at any time. Additionally, the RSG Licensing Report is also available for audit at Westinghouse offices. Therefore, the NRC determined these items would not require docketing for support of the 1.4 percent MUR power uprate review.

Plant Systems Questions

1. **On page 4.2-6 of Attachment 3 to the application, the licensee states the following: "The Westinghouse original sizing criterion recommended that the steam dump system be capable of discharging 85 percent of the rated steam flow at full-load steam pressure to permit the NSSS [Nuclear Steam Supply System] to withstand an external load reduction of up to 100 percent of plant rated electrical load without a reactor trip....For the power uprate, the large load rejection (LLR) capability was demonstrated to be 50 percent of plant-rated power without a reactor trip." The Kewaunee Updated Safety Analysis Report (USAR), page 10.1-1, states the following: "The Reactor Coolant System can accept a complete loss of external load from full power without reactor trip. The steam dump and turbine electro-hydraulic control systems make it possible to accept a full load rejection to auxiliary load using atmospheric and condenser dump without reactor or turbine trip."**

   Justify the difference between the original design and the USAR "to withstand an external load reduction of up to 100 percent of plant rated electrical load without a reactor trip" and the power uprate "50 percent of plant-rated power without a reactor trip." Explain the Kewaunee licensing basis.

   NMC Response:

   Margin-to-trip evaluations and analyses at uprated conditions of 1772 MWt indicated that the plant control systems could not support a 100 percent load rejection. However, analysis showed the plant control systems could withstand a 50 percent load rejection without a plant trip. The 50 percent load rejection analysis results indicated that adequate margin remains to the KNPP TS reactor trip setpoints. Therefore, the plant response remains acceptable. Although the 50 percent load reject capability is less than the 100 percent value currently addressed in the KNPP USAR, the 50 percent load reject capability bounds the credible events the plant could be expected to incur.

   Currently, there is a discrepancy between the 2002 update of the KNPP USAR and the MUR power uprate submittal. This is because the USAR change has not been processed. The USAR changes for the sections containing discussion on the 100 percent load rejection have been identified as part of a design modification package for the fuel transition. The changes associated with that modification are currently scheduled to be incorporated into the next planned KNPP USAR revision occurring approximately six months following the completion of the refueling outage that began on April 5, 2003.

2. Page 4.2-10 of Attachment 3 to the application states the following: "...the minimum usable inventory should be increased from 39,000 gallons to 41,500 gallons to meet the loss-of-AC [alternating current]-power licensing basis for the range of NSSS operating conditions upon approval of the 7.4% power uprate. The current capacity of the 39,000 gallons is acceptable for the 1.4% MUR power uprate."

Explain and justify the conclusion that 39,000 gallons are acceptable for the proposed 1.4-percent MUR power uprate.

NMC Response:

Attachment 3 of reference 1 contained system and component analyses performed for a 7.4 percent power uprate to 1772 MWt. For the 7.4 percent power uprate, the condensate storage tank (CST) volume must be changed to the 41,500 gallon value to support the four hour coping period for the station blackout (SBO) analysis. However, for the 1.4 percent MUR power uprate, the 39,000 gallon volume remains acceptable since the current SBO analysis contained a two percent power measurement uncertainty. The SBO analysis that bounds the MUR power uprate is described in Attachment 2 of reference 1. This item is located on page 46 of Attachment 2. The discussion from reference 1, Attachment 2, is repeated below for review purposes.

The only potential impact of the 1.4 percent MUR power uprate on the ability of the plant to withstand and recover from a station blackout (SBO) is the increased decay heat that must be removed from the RCS. The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pump, supplied with steam from the steam generators, to support reactor heat removal due to uprate. The Technical Specification minimum required volume in the condensate storage tanks (CST) is 39,000 gallons. This volume remains acceptable for the MUR power uprate since it is based on 102 percent of the current rated power of 1650 MWt. The TS CST volume and the assumed power level and uncertainty are described in an NRC safety evaluation dated November 20, 1990 (TAC 68558) and confirmed in a supplemental safety evaluation and an additional safety evaluation dated October 1, 1991 (TAC 68558) and November 19, 1992 (TAC M84521), respectively. The two percent uncertainty on the current core power of 1650 MWt bounds the uprate to 1673 MWt (a 1.4 percent uprate with 0.6 percent uncertainty). Therefore, the ability of the KNPP to respond to a SBO will not be altered due to the 1.4 percent MUR power uprate.

Reactor Systems Questions:

1. **Equation 5 (page 13) of Attachment 7 (WCAP-15591) to the application contains a term for reactor coolant pump heat addition (Q_P) and a term for primary system net heat losses (Q_L). These terms are introduced into the overall heat balance without consideration of their actual effect on the hot-to-cold leg enthalpy change, $h_H - h_C$. Since only the portions of these terms that affect heat addition or loss from the cold leg RTD location to the hot leg RTD location affect the temperature readings, and hence affect $h_H - h_C$, this equation is either an approximation that the licensee did not justify or an error.**

   Address the impact of this equation and any corrective actions that the licensee will take.

   NMC Response:

   In addition to the impact of heat input from the reactor core and heat removal from the steam generator, reactor coolant temperature changes around the RCS due to heat addition from the RCP and heat losses from the RCS components and auxiliary systems. Compression of the coolant increases coolant temperature by about 0.4°F in the RCP. Heat losses due to convection and conduction from the RCS components and due to charging flow reduce coolant temperature by about 0.04°F, and the steam generator removes the remaining 0.36°F to maintain steady state temperature. Although the heat losses are distributed around the RCS, the simplifying assumption was made that the heat losses are all applied at the RCP discharge, reducing the net temperature increase at the RCP to about 0.36°F.

   An analysis of the temperature distribution from the cold leg RTD to the hot leg RTD indicates that heat losses would reduce the T-hot measurement by less than 0.03°F. If T-hot were increased to correct for this small difference, the coolant Δh used to calculate RCS flow would increase by less than 0.05 percent, resulting in a small calculated RCS flow decrease (less than 0.05 percent or 44 gpm per loop). It was concluded that this correction was negligible and was not applied. Therefore, no corrective action is required. The uncertainty on the correction to T-hot would be much smaller than 0.03°F and would have an insignificant impact on the RCS flow measurement uncertainty.

   Note that the WCAP-15591 (page 17) conservatively assumes that "Net Heat Input to RCS" is 8.0 MWt. The actual net heat input was measured during the initial startup testing of the KNPP to be 7.11 MWt. This results in the calculated RCS flow rate being conservatively low by approximately 0.05 percent or 44 gpm. This essentially cancels out the negligible nonconservatism discussed in the above paragraph.

2. For the SI and RHR system analyses, the licensee states that the systems were evaluated for an uprated power of 1772 MWt.

Were these analyses previously approved by the NRC or were they conducted using methods or processes that were previously approved by the NRC? In addition, for the SI system, provide a discussion identifying and evaluating the effects of the higher power level on the system.

NMC Response:

The evaluation of the Kewaunee Safety Injection System (SIS) and Residual Heat Removal System (RHRS) fluid systems for the power uprate of 1772 MWt was conducted using the standard Westinghouse fluid system evaluation process. The analyses for these fluid systems for the KNPP power uprate were not previously approved by the NRC, however, the methods for analysis are considered approved and acceptable since these standard fluid system analysis methods have been applied and have supported numerous Westinghouse plant power uprates which have been approved by the NRC. Therefore the fluid system evaluation methods are considered acceptable for application to the KNPP power uprate since they have been previously applied to numerous other power uprates which have been approved by NRC.

The SIS is an engineered safeguard system that provides water inventory and cooling to the reactor and reactor coolant system (RCS) in the event of a design basis accident such as a main steam line break (MSLB) or a loss of coolant accident (LOCA). The higher power level of the KNPP power uprate does not impact the SIS directly since SIS is a standby engineered safety system. However the power level increase for the power uprate has the effect of increasing the post accident heat load thus creating greater demands on the SIS for required response time and water injection flow rate and duration. SIS flow rate, water volume injection capability and duration, and heat removal capability are all critical performance requirements for the SIS in the event of a design basis accident. These system safety performance requirements are affected by the power uprate since there is a greater initial and greater decay (post trip) heat load to remove due to the higher initial power level. The fluid system evaluation process determines specific fluid system capabilities in the post accident RCS conditions. The SIS is modeled using the calculated fluid system heat removal and flow injection capabilities. This SIS model is incorporated into the LOCA and MSLB design basis accident (DBA) analyses. The accident analyses for LOCA and MSLB are dynamic simulations of the DBA scenarios including the response of the SIS fluid system that is modeled as an engineered safety feature available for accident mitigation. The accident analysis simulates reactor and reactor coolant conditions subsequent to the postulated Reactor Coolant System and Main Steam System breaks. As a result of the accident analyses, the SIS system, and component criteria necessary to demonstrate compliance with regulatory requirements at the uprated power level are established. Since the results of these analyses have shown that the analysis acceptance criteria are satisfied, the safety analysis has demonstrated that SIS provides adequate safety margin at the uprated power level. The SIS is therefore evaluated at the higher power level and shown to meet all its safety and performance requirements.

3. **In Section 4.1.4.3.1 of Attachment 3 to the application, the licensee justifies increased plant cooldown times based on plant economics. This practice is unacceptable.**

Provide an alternate justification for the increased RHR cooldown times.

NMC Response:

The KNPP is designed so that hot shutdown is a safe and stable plant condition, which can be maintained for an extended period of time. Eventual achievement of cold shutdown conditions may be required for long term recovery from design bases events, however, there is no safety reason why this must be accomplished in some limited period of time. Therefore, the original plant design bases coodown period (from no load temperature of 547°F to 140°F) for refueling and maintenance was a contractual commitment based on plant economics.

With respect to normal plant operations, the KNPP Technical Specifications contain numerous limiting conditions of operation (LCOs) which dictate that, if an inoperable component is not restored to operable status within a specified period of time, cold shutdown must be achieved within 36 hours after hot shutdown is attained. For these LCOs both trains of the Residual Heat Removal System, Component Cooling Water System and Service Water System are available to accomplish cooldown to cold shutdown conditions (less than or equal to 200°F as defined by the Technical Specifications). At the current plant core rating of 1650 MWt and assuming a maximum design service water temperature of 80°F, cold shutdown conditions can be achieved within 16 hours after hot shutdown is attained. At the uprated core power of 1772 MWt the time to achieve cold shutdown would increase about 2 hours, that is from 16 hours to 18 hours. Since this time is well within the technical specification limit of 36 hours, the Residual Heat Removal System in conjunction with the Component Cooling Water System and Service Water System is adequately sized to achieve cold shutdown within the technical specification limits.

4. In Section 5.1.2.2 of Attachment 3 to the application, the licensee states that the calculated fluence projections used in the Power Uprate Program evaluation comply with Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

Describe how the fluence calculations comply with RG 1.190.

NMC Response:

A summary of regulatory positions regarding fluence calculational methods is provided in Section D of Regulatory Guide 1.190. The approach taken in the Kewaunee fluence calculations is to address each of these regulatory positions as summarized in Table 3 of this response. In each case, the provisions of Regulatory Guide 1.190 are satisfied.

In addition to the methodology comparisons provided in Table 3, comparisons of the Kewaunee calculations with dosimetry results from the four surveillance capsules withdrawn to date are given in Table 4. An examination of Table 4 shows consistent behavior for all reactions at all capsule locations. The overall M/C ratio for the entire data set is 0.99 with an associated sample standard deviation of 8.2 percent. The observed M/C ratios span a range from 0.84 to 1.17 for the individual sensor measurements. This set of M/C ratios for the Kewaunee surveillance capsules indicates that the Kewaunee fluence calculations meet the ±20 percent acceptance criterion specified in Regulatory Guide 1.190. The comparisons provided in Table 4 were used to validate the application of the neutron transport calculation to the Kewaunee reactor. The measurements were not used to modify the calculations in any way.

Table 3		
Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190		
FLUENCE CALCULATION METHODS		
Regulatory Position	**NRC Staff Position**	**Kewaunee Calculation**
1.3	**Fluence Determination.** Absolute fluence calculations, rather than extrapolated fluence measurements, must be used for the fluence determination.	Fluence evaluations are based on absolute calculations using the discrete ordinates transport method and the 2D/3D synthesis technique described in Section 1.3.4 of Regulatory Guide 1.190.
1.1.1	**Modeling Data.** The calculation modeling (geometry, materials, etc.) should be based on documented and verified plant specific data.	The calculation model is based on nominal dimensions obtained from plant specific design drawings. System operating temperatures, and hence coolant densities, are treated on a fuel cycle specific basis.
1.1.2	**Nuclear Data.** The latest version of the Evaluated Nuclear Data File (ENDF/B) should be used for determining cross-sections. Cross-section sets based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable. When the recommended cross-section data change, the effect of these changes on licensee specific methodology must be evaluated and the fluence estimates updated when the effects are significant.	Discrete ordinates calculations make use of the BUGLE-96 ENDF/B-VI based cross-section library. The BUGLE-96 library is a 67 group (47 neutron, 20 gamma ray) data set produced specifically for light water reactor (LWR) applications by the Oak Ridge National Laboratory (ORNL). ENDF/B-VI represents the latest version of the Evaluated Nuclear Data File (ENDF/B).
1.1.2	**Cross-Section Angular Representation.** In discrete ordinates transport calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed.	A P_5 scattering approximation was used in all of the cycle specific fluence calculations. This order of scattering exceeds the recommendation specified in Regulatory Position 1.1.2.

Table 3		
Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190		
FLUENCE CALCULATION METHODS		
Regulatory Position	**NRC Staff Position**	**Kewaunee Calculation**
1.1.2	**Cross-Section Group Collapsing.** The adequacy of the collapsed job library must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.	The BUGLE-96 library is itself a collapsed job library. The testing required by this Staff Position was performed by ORNL prior to the general release of the BUGLE-96 library and is included in the documentation package accompanying the library (RSIC Data Library Collection DLC-185).
1.2	**Neutron Source.** The core neutron source should account for local fuel isotopics and, where appropriate, the effects of moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.	Core power distribution data were used on an individual fuel cycle specific basis. In applying this data in the discrete ordinates analysis, the fission spectra, neutrons released per fission, and energy release per fission accounted for the presence of both uranium and plutonium fissioning isotopes based on the burnup of individual fuel assemblies.
1.2	**End-of-Life Predictions.** Predictions of the vessel end-of-life fluence should be made with a best estimate or conservative generic power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.	Projections for future operation were based on equilibrium fuel cycle data accounting for a 7.4% power uprate, i.e. to 1772 MWt. Future core designs are periodically reviewed to assure that they are bounded by the equilibrium core used for fluence projections when surveillance capsules are withdrawn and tested.

Table 3		
Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190		
FLUENCE CALCULATION METHODS		
Regulatory Position	**NRC Staff Position**	**Kewaunee Calculation**
1.3.1	**Spatial Representation.** Discrete ordinates neutron transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of ~2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40 intervals per octant.	The spatial mesh used by Westinghouse in the discrete ordinates calculations exceeds the minimum requirements for radial and azimuthal mesh specified in Staff Position 1.3.1. An S_{16} angular quadrature set was used in all of the cycle specific fluence calculations. This level of angular discretization exceeds the recommendation specified in Regulatory Position 1.3.1.
1.3.1	**Multiple Transport Calculations.** If the calculation is performed using two or more "bootstrap" calculations, the adequacy of the overlap regions must be demonstrated.	Not applicable. The "bootstrap" technique was not used in performing the discrete ordinates calculations.
1.3.2	**Point Estimates.** If the dimensions of the tally region or the definition of the average-flux region introduce a bias in the tally edit, the Monte Carlo prediction should be adjusted to eliminate the calculational bias. The average-flux region surrounding the point location should not include material boundaries or be located near reflecting, periodic, or white boundaries.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Kewaunee analysis.
1.3.2	**Statistical Tests.** The Monte Carlo estimated mean and relative error should be tested and satisfy all statistical criteria.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Kewaunee analysis.

Table 3		
Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190		
FLUENCE CALCULATION METHODS		
Regulatory Position	**NRC Staff Position**	**Kewaunee Calculation**
1.3.2	**Variance Reduction.** All variance reduction methods should be qualified by comparison with calculations performed without variance reduction.	Not applicable. This staff position pertains to Monte Carlo calculations. The discrete ordinates approach was used in the Kewaunee analysis.
1.3.3	**Capsule Modeling.** The capsule fluence is extremely sensitive to the geometrical representation of the capsule geometry and internal water region, and the adequacy of the capsule representation and mesh must be demonstrated.	The surveillance capsules and associated structures were modeled in the discrete ordinates calculation. Adequacy of the modeling was tested by comparison with dosimetry from withdrawn surveillance capsules. Results were shown to be within the 20% uncertainty requirement specified in Regulatory Guide 1.190 (see Table 4).
1.3.3	**Spectral Effects on RT_{NDT}.** In order to account for neutron spectrum dependence of RT_{NDT}, when it is extrapolated from the inside surface of the pressure vessel to the T/4 and 3T/4 vessel locations using the E > 1-MeV fluence, a spectral lead factor must be applied to the fluence for the calculation of ΔRT_{NDT}.	The current calculations for Kewaunee address this issue by including a calculation of iron atom displacement (dpa) distributions through the vessel wall. The dpa damage function accounts for the spectral shift toward lower energies with deeper penetration into the carbon steel vessel wall. In point of fact, these calculations are rarely used. In Regulatory Guide 1.99 Revision 2, an attenuation function intended to simulate the dpa distribution is provided. This built in function is usually used to determine values of RT_{NDT} at the T/4 and 3T/4 vessel locations.

Table 3		
Summary of Compliance with the Regulatory Positions Specified in Regulatory Guide 1.190		
FLUENCE CALCULATION METHODS		
Regulatory Position	**NRC Staff Position**	**Kewaunee Calculation**
1.3.5	**Cavity Calculations.** In discrete ordinates transport calculations, the adequacy of the S_8 angular quadrature used in cavity calculations must be demonstrated.	Assessments of the adequacy of both the S_8 angular quadrature and P_3 scattering cross-section assumptions for cavity calculations were performed as a part of an analytical sensitivity study. The results of this study led to the use of the P_5 S_{16} approximations for all calculations.
1.4.1, 1.4.2, 1.4.3.	**Methods Qualification.** The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytical uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytical uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.	Measurement to calculation comparisons for the PCA Pressure Vessel Simulator described in NUREG/CR-6454 and the H. B. Robinson Unit 2 Pressure Vessel Benchmark described in NUREG/CR-6453 were completed. Likewise, an analytical sensitivity study assessing the uncertainty associated with important geometric, material density, and neutron source input parameters was performed. In combination, the results of these studies establish an overall calculational uncertainty of less than 20%.
1.4.3	**Fluence Calculational Uncertainty.** The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be $\leq$20% for RT_{PTS} and RT_{NDT} determination. In these applications, if the benchmark comparisons indicate differences greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.	The qualification of the methodology demonstrates an uncertainty of less than 20% (1 sigma). None of the benchmarking comparisons exceeded the 20% criteria specified in Staff Positions 1 and 1.4.3. Therefore, no adjustments to the calculated results have been required.

Table 4				
Comparison of Measured and Calculated Sensor Reaction Rates From Kewaunee Surveillance Capsule Dosimetry				
	M/C Ratio			
Reaction	**Capsule V**	**Capsule R**	**Capsule P**	**Capsule S**
Cu-63(n,α)Co-60	1.03	0.96	0.99	1.04
Fe-54(n,p)Mn-54	1.00	1.02	1.02	0.97
Ni-58(n,p)Co-58	0.90	1.02	0.93	1.03
U-238(n,f)Cs-137 (Cd)	0.96	1.03	0.98	0.85
Np-237(n,f)Cs-137 (Cd)	1.14		1.17	0.84
Average	**1.01**	**1.01**	**1.02**	**0.94**

Note: The overall average of the 19 sample data set is 0.99 with an associated sample standard deviation of 8.2 percent.

5. **On page 60 of Attachment 2 to the application, the licensee states that the protection system settings will be rescaled for the proposed power level of 1673 MWt.**

   Provide a discussion on the changes and a justification for the changes.

   NMC Response:

   To support the 1.4 percent MUR power uprate, instrumentation calibration and scaling changes consistent with the specific power increase will be performed. The instrument changes required to support the 1.4 percent MUR power uprate (increase licensed power from 1650 MWt to 1673 MWt) are:

   - The full power Δt_0 inputs to the overtemperature delta T (OTΔT) and the overpower delta T (OPΔT) setpoints will be changed to the predicted value which is based on best estimate evaluations for the 1.4 percent uprated power level (1673 MWt) condition.
   - Gain adjustments to the power range nuclear instruments (NIs) will also be performed. The output from the NIs will be adjusted based on a secondary heat balance calculated for the new 100 percent licensed power level (1673 MWt). Once the power range NIs have been adjusted to the appropriate percent power for the new licensed power level, all the power range reactor trips, rod stops, and permissives (P-7, P-8, and P-10) that are based on percent power will function at the appropriate values.

   Adjustment of the currents for the power range axial offset instrumentation is not required. Once the new licensed power level has been achieved, the need for changes to the axial offset instrumentation will be based on flux maps performed at the new licensed core power.

The intermediate range (IR) and source range (SR) nuclear instrumentation do not require a setpoint change due to the 1.4 percent MUR power uprate. Technical Specification Table 3.3.2-1 requires the intermediate range high flux trip to actuate ≤40 percent of licensed rated power (RP). The setpoint for the intermediate range rod stop setpoint (34 percent) and reactor trip setpoint (39 percent) for each operating cycle is determined by reactor engineering prior to initial startup following refueling and used through the entire cycle. The actual setpoints are based on 90 percent of the detector currents measured during the previous cycle shutdown and are corrected for predicted changes associated with the new core. Correcting the setpoints to account for the 1.4 percent MUR would result in an increase of the rod stop and reactor trip setpoints by approximately 0.42 percent and 0.5 percent, respectively. Since the intermediate range rod stop and trip are provided as startup protection backup to the power range instruments and are blocked above permissive P-10 (approximately 9 percent RP), using the preexisting intermediate range settings is considered conservative.

References:

1. Letter NRC-03-004 from Thomas Coutu to Document Control Desk, "License Amendment Request 193, Measurement Uncertainty Recapture power Uprate for Kewaunee Nuclear Power Plant," dated January 13, 2003 (TAC No. MB7225).

2. Letter from C.R. Steinhardt (WPSC) to US NRC Document Control Desk, "Response to NRC Bulletin No. 88-11," November 14, 1991.

3. Letter from A.G. Hansen (NRC) to C.A. Schrock (WPSC), "Kewaunee Nuclear Power Plant: Leak-Before-Break Evaluation of Pressurizer Surge Line (TAC. No. M72140)," January 3, 1992.

ATTACHMENT 2

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Excerpt (Section 8.4.2) from WCAP-16040-NP, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report," Non-Proprietary, February 2003

Water Systems

- Service Water
- Component Cooling Water

Emergency Core Cooling Systems

- Containment Spray
- Safety Injection

8.4.2 Description of Evaluation and Analysis

System operation at power uprate conditions generally results in increased pipe stress levels and pipe support loads due to slightly higher operating temperatures, pressures, and flow rates internal to the piping. The piping systems affected by the power uprate conditions were evaluated as follows:

Pre-power uprate and power uprate system operating data (operating temperature, pressure, and flow rate) were obtained from heat balance diagrams, calculations, and/or other applicable reference documents.

Change factors were determined, as required, to evaluate and compare the changes in operating conditions. The thermal, pressure and flow rate change factors were based on the following ratios:

- The thermal change factor was based on the ratio of the power uprate to pre-power uprate operating temperature. That is, thermal change factor is $(T_{\text{power uprate}}-70°F)/(T_{\text{pre-power uprate}}-70°F)$.
- The pressure "change factor" was determined by the ratio of $(P_{\text{power uprate}}/P_{\text{pre-power uprate}})$.
- The flow rate "change factor" was determined by the ratio of $(Flow_{\text{power uprate}}/Flow_{\text{pre-PowerUprate}})$.

These thermal, pressure, and flow rate change factors were used in determining the acceptability of piping systems for power uprate conditions.

Based on the thermal, pressure, and flow rate change factors determined as described above, the following engineering activities were performed and/or conclusions reached.

For thermal, pressure, and flow rate change factors less than or equal to 1.0 (that is, the pre-power uprate condition envelops or equals the power uprate condition), the piping system was concluded to be acceptable for power uprate conditions.

For thermal, pressure, and flow rate change factors greater than 1.0 through 1.05 (that is, a greater than zero and less than or equal to 5-percent increase in thermal expansion, pressure, and/or flow rate effects), this minor increase was concluded to be acceptable based on the following rationale. Certain levels of deviation from design basis conditions can be concluded to be permissible if that level of change would not alter the piping system results to an appreciable degree. Relatively small temperature changes can be concluded to be acceptable as the increase in pipe stresses, including stress levels used to postulate pipe break locations, pipe support loads, nozzle loads, jet impingement forces, pipe break loads, pipe whip restraint loads, and piping displacements, are correspondingly small and generally predictable. These increases are somewhat offset by conservatism in analytical methods used to calculate thermal and/or fluid transient stresses and loads. Conservatism may include the enveloping of multiple thermal operating conditions, as well as not considering pipe support gaps in thermal analyses. Also, for supports installed on safety related systems which are evaluated for seismic loading effects, a potential 5-percent increase in a specific thermal loading condition will generally result in a less than 5-percent overall pipe support design load increase due to the existence of seismic earthquake loads.

For thermal, pressure, and flow rate change factors greater than 1.05, more detailed evaluations were performed to address the specific increase in temperature, pressure and/or flow rate in order to document design basis compliance. Descriptions of the evaluations performed are provided in the following individual piping system sections.

8.4.2.1 Steam and Power Conversion Systems

8.4.2.1.1 Main Steam

A review of the power uprate data indicates that the operating temperature, pressure and flow rate will be increasing at 100-percent power conditions. A summary of these increases, including corresponding thermal, pressure and flow rate change factors follows:

Table 8.4.2-1 MSS Pre-Uprate and Power Uprate Operating Data			
Outlet of Steam Generator	**Pre-Uprate**	**Power Uprate**	**Change Factor**
Temperature °F	509	518	1.02
Pressure psia	741	797	1.08
Flow Rate lb/hr	7.13M	7.75M	1.09

Although the temperature and pressure have increased based on the heat balance data, the existing main steam (MS) pipe stress analyses considered higher bounding temperatures and pressures (that is, 550°F and 1100 psig) in the piping analyses. Since the operating temperature and pressure associated with power uprate are bounded by the corresponding values in the piping analyses, these temperatures and pressures were concluded to be acceptable.

The main concern for the MS piping was related to the flow rate increase and its impact on the determination of fluid transient loads resulting from a turbine-stop-valve closure event.

A detailed assessment of the MS piping and support system from the steam generators to the turbine stop valves was performed to evaluate the higher flow rate resulting from power uprate conditions. The results of this analysis concluded that the existing main steam piping system remains acceptable for power uprate conditions.

8.4.2.1.2 Bleed Steam

The existing operating temperature, pressure, and flow rate will be increasing as a result of power uprate. A summary of the temperature and pressure increases including applicable change factors is provided in Table 8.4.2-2.

Table 8.4.2-2 Bleed Steam System Pre-Uprate and Power Uprate Operating Data				
System Boundary	**Operating Parameter**	**Pre-Power Uprate**	**Power Uprate**	**Change Factor**
Extraction at Inlet of Feedwater Heaters 11A&B	Temperature (°F)	158	162	1.05
	Pressure (psia)	4	5	1.25
Extraction at Inlet of Feedwater Heaters 12A&B	Temperature (°F)	215	219	1.03
	Pressure (psia)	15	16	1.07
Extraction at Inlet of Feedwater Heaters 13A&B	Temperature (°F)	281	287	1.03
	Pressure (psia)	47	51	1.09
Extraction at Inlet of Feedwater Heaters 14A&B	Temperature (°F)	362	368	1.02
	Pressure (psia)	157	169	1.08
Extraction at Inlet of Feedwater Heaters 15A&B	Temperature (°F)	436	443	1.02
	Pressure (psia)	364	394	1.08

As shown above, the resulting thermal change factors for the extraction piping are less than or equal to the 1.05 acceptance limit. Hence, the power uprate temperatures indicated above are concluded to be acceptable.

The pressure data summarized above results in change factors greater than 1.05. However, the actual pressure and piping pressure stress increases are not significant. Therefore, pressure increases summarized above are concluded to be acceptable.

Extraction steam line flow rate increases vary from 9 to 11 percent. There are no specific fluid transient analyses that have been considered in the existing qualification of the Extraction Steam System, and historically this system does not experience severe flow induced fluid transients. Therefore, the flow rate increases for this piping are concluded to be acceptable.

8.4.2.1.3 Condensate

A review of the power uprate data for the Condensate System reveals that the existing operating temperature and flow rate will be increasing as a result of power uprate. The operating pressure of the Condensate System will remain unchanged at 400 psia as a result of power uprate.

A summary of the Condensate System temperature increases, including "change factors," is provided in Table 8.4.2-3.

Table 8.4.2-3

Condensate System Pre-Uprate and Power Uprate Operating Data

System Boundary	Operating Parameter	Pre-Power Uprate	Power Uprate	Change Factor
Condensate Pump to Heaters 11A&B	Temperature (°F)	90	95	1.25
Heaters 11A&B to 12A&B	Temperature (°F)	152	156	1.05
Heaters 12A&B to 13A&B	Temperature (°F)	210	214	1.03
Heaters 13A&B to 14A&B	Temperature (°F)	273	278	1.02
Heaters 14A&B to Feedwater Pumps	Temperature (°F)	360	366	1.02

As shown above, for the piping between the condensate pump and FW heaters 11A and B, the resulting thermal change factor of 1.25 is based on the temperature increasing from 90°F to 95°F. Since the temperature increase is limited to only 5°F, and the resulting 95°F value is considered a low temperature with respect to piping qualification concerns, this portion of the Condensate System is considered acceptable for power uprate operating conditions.

The resulting thermal change factors for the balance of the Condensate System are less than or equal to the 1.05 acceptance limit. Hence, the power uprate temperatures indicated above are concluded to be acceptable.

The flow rate through the condensate pumps will be increasing by approximately 8 percent from 4.98M lb/hr to 5.39M lb/hr. Since the Condensate System does not contain any fast closing valves, no previous flow-induced fluid transient events have been identified, and no specific fluid transient evaluations/analyses have been performed on this piping, the subject flow rate increase is concluded to be acceptable.

8.4.2.1.4 Feedwater

A review of the power uprate data for the AFS reveals that the existing operating temperature and flow rate will be increasing as a result of power uprate. The operating pressure of the AFS will remain unchanged at 1200 psia as a result of power uprate. A summary of the temperature and pressure data including applicable change factors is provided in Table 8.4.2-4.

Table 8.4.2-4

FW System Pre-Uprate and Power Uprate Operating Data

System Boundary	Operating Parameter	Pre-Power Uprate	Power Uprate	Change Factor
FW Pump Discharge to Inlet of FW Heater	Temperature (°F)	362	368	1.02
	Pressure (psia)	1200	1200	1.0
Outlet of FW Heater to Steam Generators	Temperature (°F)	430	437	1.01
	Pressure (psia)	1200	1200	1.0

As shown above, the maximum temperature increase is only 7°F, and the resulting thermal change factors are less than the 1.05 acceptance limit. Hence, the power uprate temperatures summarized above are concluded to be acceptable.

The pressure change factors are 1.0 and are also concluded to be acceptable.

Since the AFS does not contain any fast closing valves, and no previous flow induced fluid transient events have been identified, and no specific fluid transient analyses have been performed on this piping, the subject flow rate increase is concluded to be acceptable.

ATTACHMENT 4

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Westinghouse Tables for Support of WCAP-15591, Revision 1 (Non-proprietary)

Kewaunee Pressurizer Pressure Control Uncertainty for 1780 MWt-NSSS Power

+a,c

Transmitter Uncertainties

Protection Rack Uncertainties

Control Board Indicator Uncertainties

WCAP-15591, Rev 1
(in support of Table 1)

Kewaunee Pressurizer Pressure Control Uncertainty for 1780 MWt-NSSS Power

+a,c

Controller Uncertainties

Electronics Uncertainty

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

Feedwater Flow (Differential Pressure) on feedwater bypass loop

Feedwater Pressure on feedwater bypass loop

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

Feedwater Temperature

WCAP-15591, Rev 1
(in support of Table 3a)

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

Steam Pressure

WCAP-15591, Rev 1
(in support of Table 3a)

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

RCS Hot leg Temperature

rcsflowcal_venturi_computer02_Case

WCAP-15591, Rev 1
(in support of Table 3a)

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

RCS Cold leg Temperature

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

Pressurizer Pressure

Transmitter Uncertainties

Protection Rack Uncertainties

Computer Indication Uncertainties

Kewaunee Calorimetric RCS Flow Uncertainty for 1757 MWt - NSSS Power

+a,c

Pressurizer Pressure (continued)

WCAP-15591, Rev 1
(in support of Table 7)

Kewaunee Calorimetric Power Measurement Uncertainty for 1757 MWt-NSSS Power with Venturis

Feedwater Temperature

+a,c

WCAP-15591, Rev 1
(in support of Table 7)

Kewaunee Calorimetric Power Measurement Uncertainty for 1757 MWt-NSSS Power with Venturis

Feedwater Flow (Differential Pressure)

+a,c

WCAP-15591, Rev 1
(in support of Table 7)

Kewaunee Calorimetric Power Measurement Uncertainty for 1757 MWt-NSSS Power with Venturis

Steam Pressure

+a,c

Pwrcal_venturi_computer02_Case

WCAP-15591, Rev 1
(in support of Table 10)

Kewaunee Calorimetric Power Measurement Uncertainty for 1780 MWt-NSSS Power with UFMs and UTMs

Feedwater Flow

[ ] + a, c

Feedwater Temperature

[ ] + a, c

Kewaunee Calorimetric Power Measurement Uncertainty for 1780 MWt-NSSS Power with UFMs and UTMs

Steam Pressure

+a,c

Kewaunee Calorimetric Power Measurement Uncertainty for 1780 MWt-NSSS Power with UFMs and UTMs

Steam Generator Blowdown Loop Flow

+a,c

WCAP-15591, Rev 1
(in support of Table 13)

Kewaunee Calorimetric Power Measurement Uncertainty for 1777 MWt-NSSS Power with UFMs and RTDs

Feedwater Flow

+a,c

Feedwater Temperature

+a,c

WCAP-15591, Rev 1
(in support of Table 13)

Kewaunee Calorimetric Power Measurement Uncertainty for 1777 MWt-NSSS Power with UFMs and RTDs

Steam Pressure

+a, c

WCAP-15591, Rev 1
(in support of Table 13)

Kewaunee Calorimetric Power Measurement Uncertainty for 1777 MWt-NSSS Power with UFMs and RTDs

Steam Generator Blowdown Loop Flow

+a, c

ATTACHMENT 5

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Westinghouse Authorization Letter, CAW-03-1632, Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 3

Westinghouse

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

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

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

Our ref: CAW-03-1632

April 25, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Information Required to Support Response to Request for Additional Information (RAI) from NRC I&C Branch on Kewaunee 1.4% Power Uprate" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-03-1632 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 (NMC).

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1632 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: S. J. Collins
D. Holland
B. Benney

CAW-03-1632

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 25th day
of April, 2003

Sharon L. Fiori

Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, 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 rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 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 several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

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

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, 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 that 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 and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in letter dated April 25, 2003, "Information Required to Support Response to Request for Additional Information (RAI) from NRC I&C Branch on Kewaunee 1.4% Power Uprate" (Proprietary), April, 2003, being transmitted by the Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse Electric Company LLC for Kewaunee Nuclear Plant is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Revised Thermal Design Procedure Instrument Uncertainty Methodology.

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

(a) Provide documentation of the analysis and methods for determining operating parameter uncertainties.

(b) Calculate information which is used in thermal analysis of the nuclear fuel.

(c) Assist the customer in obtaining 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.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

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 calculations 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.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) 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 10 CFR 2.790(b)(1).

COPYRIGHT NOTICE

ATTACHMENT 6

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

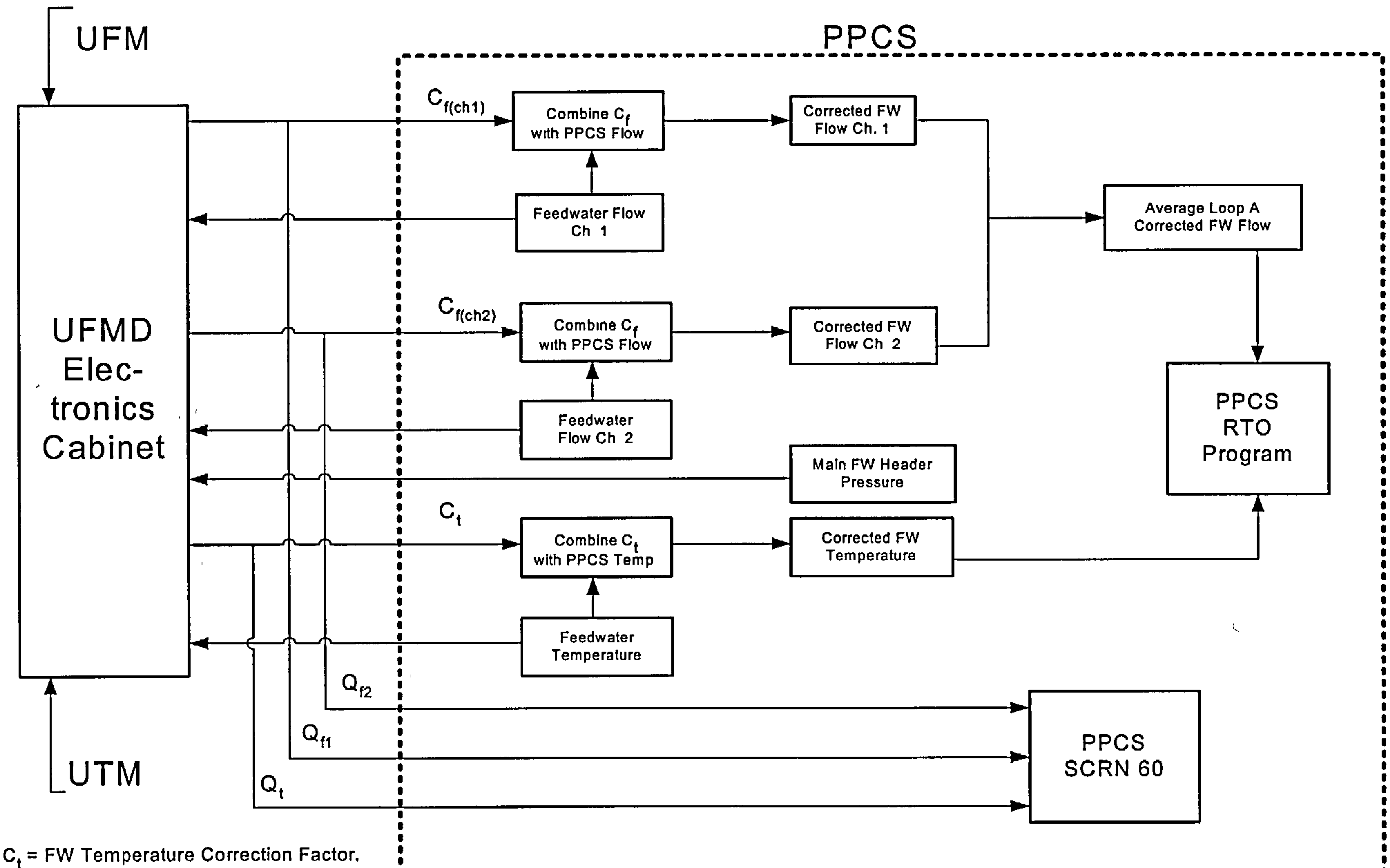
Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Figures Supporting the I&C Responses
(Figure 1, Figure 2, and Drawing M-205)

Docket 50-305
NRC-03-047
April 30, 2003
Attachment 6, Page 1

Feedwater Loop A Data Point Processing and Communication Links



C_t = FW Temperature Correction Factor.
C_f = FW Flow Correction Factor.
Q_t = Quality Signal For UFMD Temperature.
Q_f = Quality Signal For UFMD Feedwater Flow.

Figure 1

PRIMARY OPERATOR PPCS SCREEN 60

UFMD / UTM CORRECTION FACTORS			
LOOP A		**LOOP B**	
FT-466 CORRECTION	X.XXXX	FT-476 CORRECTION	X.XXXX
FT-467 CORRECTION	X.XXXX	FT-477 CORRECTION	X.XXXX
TE-15043 CORRECTION	X.XXXX	TE-15044 CORRECTION	X.XXXX

UFMD OPERATING LIMIT	XXXX MWt
RTO OPERATING LIMIT	XXXX MWt
CURRENT REACTOR OUTPUT	XXXX MWt
	<15 MINUTE SLIDING AVG>

[APPLY UFMD LIMIT]

Note the above screen structure may change as the KNPP operators obtain experience with the system.

Figure 2

ATTACHMENT 8

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Responses to Kewaunee CROSSFLOW RAIs (Non-proprietary)

Westinghouse Electric Company
2000 Day Hill Road
Windsor, CT 06095

3. The submittal proposes the use of ultrasonic temperature measurements. As mentioned in attachment 2 (I.1), these are "...not described in CENPD-397-P-A,". In attachment 7, figure 3, Calorimetric Power Measurement (Using ..., the figure indicates use of the UTM for density correction and enthalpy of the feedwater. Furthermore, attachment 2, table I.1 (page 9) suggests an additional 0.2 % reactor thermal power to be gained by using the UTMs. Please provide information on the use of the UTMs in sufficient detail for the staff to evaluate its use. The following are some items to include:

- **The type of sensor and the theory of operation**

The UTM® Ultrasonic temperature measurement system, CORRTEMP, uses clamp-on ultrasonic transducers to measure the feedwater temperature. [

]+a,c

Knowledge of the acoustical velocity and the pressure of the feedwater, which is another input to the UTM, defines the thermodynamic state of the feedwater. Once the thermodynamic state of the feedwater is known, it is then possible to determine the corresponding temperature of the feedwater using the NIST water and steam property tables.

[

]+a,c

Westinghouse Electric Company
2000 Day Hill Road
Windsor, CT 06095

- **Is it an analog or digital sensor?**

The ultrasonic signal that passes between the two transducers and the determination of the time delay are all considered to be analog processes. However, the time delay is then converted to a digital signal, so that all the calculations associated with the determination of the feedwater temperature can be done digitally.

- **A diagram of the sensor (preferably from the vendor drawings)**

+a,c

- **Make, model number etc**

From the packing slip....

CORRTEMP Components	Model Numbers
Multiplexer	SMX–2160
Transducer	TPB – 1000
UTM Frame	TMF – 1016
Signal Conditioning and Processing Unit	SCP - 1000

- **How were the sensitivity and uncertainty values determined (i.e. where do the numbers come from)**

Westinghouse Electric Company
2000 Day Hill Road
Windsor, CT 06095

[

] +a,c

- **How does the UTM interfaces with the plant computer and have those interfaces been considered in the uncertainty determination.**

  Since the communication between the UTM and the plant computer is digital, no added uncertainty is introduced when information is passed between the UTM and the plant computer.

- **How will the UTM's be calibrated and what is the suggested calibration periodicity.**

  [

  ] +a,c

  In addition to these external checks, the UTM also includes many internal performance checks that will block the data generation and alarm Operations, if the UTM performance degrades.

5. **Section 8.1.3 of CENPD-397-P-A discusses the transducer installation. Please discuss any expected differences in M/TSF support frame area temperature from the time it is measured at installation and the time KNPP will be operating.**

   At KNPP the UFMD installation and calibration were performed at full power with nominal feedwater temperatures.

   [

   ] +a,c

Westinghouse Electric Company
2000 Day Hill Road
Windsor, CT 06095

Changes in the transducer spacing and the cross-sectional flow area due to changes in feedwater temperature are automatically accounted for by the UFMD computer. [

] +a,c

ATTACHMENT 9

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

April 30, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to Requests for Additional Information Regarding LAR 193

Westinghouse Authorization Letter, CAW-03-1633, Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice for Attachment 7

Westinghouse

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

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

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

Our ref. CAW-03-1633

April 28, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Response to NRC Request for Additional Information (RAI) on Kewaunee Crossflow Ultrasonic Flow Measurement System"

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-03-1633 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 (NMC).

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1633 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: S. J. Collins
D. Holland
B. Benney

CAW-03-1633

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

H. A. Sepp

H. A. Sepp, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 28th day
of April, 2003

Sharon L. Fiori
Notary Public

SHARON L. FIORI COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

Westinghouse

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

Mr. Harv Hanneman, Project Manager
Nuclear Management Company LLC
Kewaunee Nuclear Power Plant
N49, State Highway 42
Kewaunee, WI 54216-9511

Direct tel: (412) 374-4037
Direct fax: (412) 374-4011
e-mail: owocrh@westinghouse.com

Ref: LTR-IPES-03-81

April 28, 2003

Nuclear Management Company LLC
Kewaunee Nuclear Power Plant
Measurement Uncertainty Recapture Power Uprate
Response to NRC RAIs on Crossflow Ultrasonic Flow Measurement System

Dear Mr. Hanneman:

This letter transmits 2 copies of proprietary "Response to NRC Request for Additional Information (RAI) on Kewaunee Crossflow Ultrasonic Flow Measurement System" and 2 copies of nonproprietary versions of "Response to NRC Request for Additional Information (RAI) on Kewaunee Crossflow Ultrasonic Flow Measurement System," April 28, 2003, for your submittal to the NRC for review and approval.

In addition to the proprietary and nonproprietary information, there are four other enclosures for your use:

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-03-1633) with Affidavit CAW-03-1633.

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

If you have any questions, please do not hesitate to contact us.

Very truly yours,

Patrick R. Kottas
Crossflow Project Manager

R. H. Owoc
Power Uprate Project Manager

Enclosures

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, 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 rule making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 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 several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

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

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, 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 that 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 and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in letter dated April 28, 2003, "Response to NRC Request for Additional Information (RAI) on Kewaunee Crossflow Ultrasonic Flow Measurement System" (Proprietary) being transmitted by the Nuclear Management Company LLC letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse Electric Company LLC for Kewaunee Nuclear Plant is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of plant operation with the Crossflow Measurement System.

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

(a) Provide documentation in support of implementation of an ultrasonic feedwater flow meter.

(b) Provide methodology or analysis in support of a Crossflow Flow Measurement System.

(c) Assist the customer in the licensing process.

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.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

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 methodology 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.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) 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 10 CFR 2.790(b)(1).

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