January 21, 2003

Mr. Thomas Coutu
Site Vice President and Interim Plant Manager
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 State Highway 42
Kewaunee, WI 54216

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT REQUEST CONFORMING TECHNICAL SPECIFICATION CHANGES FOR USE OF WESTINGHOUSE VANTAGE+ FUEL (TAC NO. MB5718)

By letter dated July 26, 2002, Nuclear Management Company, LLC (NMC or the licensee) submitted a request for a proposed amendment to revise Kewaunee Nuclear Power Plant technical specifications for use of Westinghouse VANTAGE + fuel.

The Nuclear Regulatory Commission (NRC) staff finds that the additional information identified in Enclosure 1 is needed.

A draft of the request for additional information was e-mailed to Mr. T. Maloney (NMC) on November 21 and 25, 2002, and G. Riste (NMC) on January 13, 2003.

A meeting was held between NMC, Westinghouse, and NRC on January 15, 2003, to discuss the questions to ensure that there was no misunderstanding. Enclosure 2 has a list of the meeting attendees. Also, the meeting established a mutually agreeable response date within 30 days from the date of this letter.

Please contact me at (301) 415-1446 if future circumstances should require a change in this response date.

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Request for Additional Information
2. Meeting attendee list

cc w/encl: See next page

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KEWAUNEE NUCLEAR POWER PLANT

REQUEST FOR ADDITIONAL INFORMATION

REGARDING TECHNICAL SPECIFICATION CHANGES

FOR USE OF WESTINGHOUSE VANTAGE PLUS FUEL

DOCKET NO. 50-305

Attachment 1

1. Page 3 of Attachment 1 lists a number of proposed technical specification (TS) changes associated with this license amendment request. However, the marked up TS pages are not included in the package. Please provide the marked up TS pages.

2. Page 4 of Attachment 1 refers to References 1-17, 1-18 and 1-19, and Reference 5, Table 5.1-8. We are unable to locate these references in the amendment package. Please provide us with clearer information regarding these references.

3. Page 5 of Attachment 1 states that, “Empirical data acquired during Cycle 25 confirms that this fuel is both compatible with Kewaunee Nuclear Power Plant (KNPP) reactor design and with the Framatome/aircraft nuclear propulsion (ANP) fuel currently in use.” Please discuss the data acquired and technical basis for reaching this conclusion.

Attachment 2

1. The licensee is proposing to revise the TS Bases Section 2.1, “Safety Limits - Reactor Core,” to include three departure from nucleate boiling ratio (DNBR) correlations used in the safety analyses. These three correlations are WRB-1, HTP and W-3. This is not consistent with TS 2.1.b, which provides DNBR limits for only WRB-1 and HTP DNB correlations. Why is the W-3 correlation and its associated limit not specified in the proposed revision to TS 2.1? Why isn’t the statistical methodology (RTDP) incorporated into the DNBR safety limit or discussed in the proposed TS Bases section?

2. TS Sections 2.3.a.3.A and 2.3.a.3.B specify the Overtemperature and Overpower ΔT trip setpoint equations. The proposed changes include a change in the definition of the variable T. The current KNPP TS define T as the Average temperature. The proposed TS change defines T as the Reference Average Temperature at Rated Power. Please provide the technical basis for this change. This change is not consistent with WCAP-8745-P-A, “Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,” or the Westinghouse Standard Technical Specifications.

3. The licensee has proposed many changes to TS Section 3.10 - Control Rod and Power Distribution Limits. The licensee seems to be adopting portions of NUREG-1431, “Standard Technical Specifications (STS) Westinghouse Plants” while maintaining portions of the existing KNPP TS. With respect to the proposed changes to TS Section 3.10:

a. Please provide technical justification for the deviation from STS action completion times.

b. STS for $F_Q^N(z)$ and $F_{\Delta H}^N(z)$ contain an action to eventually reduce to MODE 2 ($\leq$5 percent rated thermal power (RTP)) if required actions and associated completion times cannot be satisfied. The proposed KNPP TSs do not include such an action statement. Please provide justification for this.

c. The proposed KNPP TS changes incorporate a change in methodology from Constant Axial Offset Control (CAOC) to Relaxed Axial Offset Control (RAOC). The KNPP submital does not discuss this change in methodology. Please discuss the purpose and basis for this change in methodology, provide a reference to the Nuclear Regulatory Commission (NRC) approved methodology being applied and discuss the validity of this methodology for KNPP?

d. Proposed TS 3.10.b.5 requires that the measured $F_Q^{EQ}(z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80 percent of the core. Considering the new V422+ fuel and uprated power conditions, does the 80 percent value remain adequate?

e. Proposed TS 3.10.b.6.B has a typo, the last portions should read as "in accordance with TS 3.10.b.6.A."

f. Proposed TS 3.10.b.6.C refers to a peak pin power parameter of interest as F_Q^N. The current KNPP TS, which is similarly worded, refers to the peak pin power parameter as $F_{\Delta H}^N$. The proposed TS wording is identical to the current TS except that this parameter has changed from F_Q^N to $F_{\Delta H}^N$. Please provide the technical basis for this proposed change.

g. Proposed TS 3.10.b.6.C provides the actions addressing an increase of F_Q^N by 2 percent or more, when compared to the last power distribution map. Item 2.6.3 of the Draft KNPP core operating limit report (COLR) states, "The penalty factor for TS 3.10.b.5.C.i (which appears to be a typo) shall be 2 percent." Please discuss the magnitude and method of the penalty applied and actions if the increase is greater than 2 percent.

h. Proposed TS 3.10.b.8 provides the requirements regarding Axial Flux Difference (AFD) Limits. Please provide the technical bases for allowing 15 minutes to restore AFD to within limits prior to reducing power to less than 50 percent RTP and for TS 3.10.8.B, which provides the actions and completion times for the AFD alarms being inoperable. These actions are not included in STS or KNPP current TS.

i. KNPP has adopted much of the STS Section 3.2 Power Distribution Limits. Please discuss why KNPP is not revising the current TS for Quadrant Power Tilt in accordance with the STS.

j. Proposed TS 3.10.m changes the Reactor Coolant flow from a loop flow of $\geq$93,000 gpm to a total flow rate of $\geq$178,000 gpm. Please provide the technical basis for this change.

Attachment 3 - Westinghouse Report - Technical Basis for Transitioning to V422+ Fuel

1. The licensees submittal references the NRC staff's Safety Evaluation Report (SER) regarding the acceptability of VANTAGE+ fuel (WCAP-12610-P-A). With respect to oxidation and crud buildup, the staff's conclusions were based on a limited amount of available ZIRLO oxidation data. As such, page 9 of the SER includes the requirement that, "Future plant applications of VANTAGE+ design must also demonstrate that the ZIRLO oxidation data is applicable for these applications." Please discuss the review performed to ensure this data is acceptable for application at KNPP.

2. Section 1.6 - The licensee states that the reactor trip setpoints for the new analyses performed assuming the uprated power will also be bounding because they were generated for the uprated power, but will be used for the current nuclear steam supply system (NSSS) power level. The licensee concludes that they are therefore, more restrictive than the current trip setpoints that are being used. This statement implies that reactor trip setpoints are being revised, which requires a TS change. Which setpoints are being changed as part of this license amendment request? Also, please provide the technical basis for the conclusion that all reactor trip setpoints used in the new uprate power analyses are more restrictive than the current setpoints.

3. To calculate the uncertainties for the RTDP methodology, the analyses in WCAP-15591 (the KNPP specific RTDP analysis) assumed an uprated power of 1757 MWt (NSSS) and 54F replacement steam generators. The licensees submittal proposes an uprated power level of 1780 MWt (NSSS). Please quantify any differences in the WCAP-15591 RTDP analysis results for an uprate power of 1780 MWt. What impact does this have on the DNBR limits?

4. Table 1-2 in the Westinghouse report specifies an reactor coolant system (RCS) design flow of 89,000 gpm/loop. The licensee has proposed a new updated final safety analysis report (UFSAR) Table 14.0.3 which provides a table of nominal values for non-loss-of-coolant accident (LOCA) analyses for RTDP methodology, and applies uncertainties to these nominal values for STDP methodology. However, the way that the RCS flow value from Table 1-2 is used is not consistent with the way that the Tavg and RCS pressure values are used. Please discuss the technical basis for using 93,000 gpm value in this table. Why is uncertainty not applied to the 89,000 gpm nominal value for the STDP methodology? When comparing Table 1-2 of the Westinghouse report with new USFAR Table 14.0.3, it appears that a non-conservative RCS flow value may have been used in the safety analyses. Please discuss this discrepancy and how it relates to the proposed minimum RCS flow TS value.

5. The NRC staff issued the SER for WCAP-11397-P-A, "Revised Thermal Design Procedure," in January 1989. Please provide a discussion which demonstrates how each of the seven restrictions outlined in the staff's SER for this methodology are satisfied considering the proposed new fuel to be used at KNPP.

6. The proposed new V422+ fuel design for KNPP incorporates VANTAGE + features, PERFORMANCE + features and other KNPP specific features (0.422 inch outside diameter fuel rod and instrumentation tube and new optimized fuel assembly (OFA) style mid-grid). The NRC staff has reviewed the VANTAGE + features in WCAP-12610-P-A. The PERFORMANCE + features were evaluated by Westinghouse in SECL-92-305 under 10 CFR 50.59 guidelines. Please explain why the PERFORMANCE + features were not evaluated under the NRC approved Fuel Criteria Evaluation Process ?

7. The Westinghouse report states that the 422V+ and the 14x14 Framatome/ANP designs are mechanically and hydraulically compatible with each other. Westinghouse bases this conclusion on a referenced report (PD2-01-46, Rev. 1, "Kewaunee Fuel Transition Work Report, Revision 1 to Fuel Assembly Compatibility Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assembly," November 1, 2001). Please list the technical considerations (items) that were evaluated in this referenced report and discuss the methodologies applied. Please include any references to NRC-approved methodologies.

8. Regarding compatibility of the 422V+ and Framatome/ANP fuels with respect to crossflow, Westinghouse provides a criterion that there should be outer grid strap overlap between any two fuel assemblies in the core throughout their life in the core. Please provide further clarification regarding this statement as follows:

    a. What is meant by outer grid strap overlap and the quantity of overlap required?

    b. The technical basis for this criteria?

    c. A reference to the NRC-approved methodology applied to determine that this criteria is satisfied for a core consisting of 422V+ and Framatome/ANP fuel.

9. Section 2.2.3 of the Westinghouse report states that, "The diameter of the 422V+ guide thimbles is 0.015 inches smaller than that of the 14x14 Framatome/ANP design. The 422V+ guide thimble inner diameter provides a minimum diametral clearance of 0.0088 inches (under worst case conditions) for control rods supplied by Westinghouse." What are the accepted design criteria established for minimum clearance, and please define the worst case conditions. Is rod cluster control assembly (RCCA) insertion time impacted by this change?

10. Section 2.4.1 of the Westinghouse report addresses fuel rod design criteria. The report addresses a list of various criteria, but does not address all criteria listed in NUREG-0800, Standard Review Plan (SRP) Section 4.2. Please provide the basis for choosing only the criteria evaluated in Section 2.4.1. Provide additional evaluations as necessary.

11. In Section 2.4.1 of the Westinghouse report regarding rod internal pressure, it is stated that, "The design limit for Condition II events is that DNB propagation is not extensive, that is, the process is shown to be self-limiting and the number of additional rods in DNB due to propagation is relatively small." NUREG-0800, SRP Section 4.2 specifies the acceptance criteria for evaluation of fuel design limits. One of the criteria provides

assurance that there be at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience DNB during normal operation or anticipated operational occurrences (Condition II events). Does this contradiction impact the conclusions reached in the Westinghouse report?

12. In Section 2.4.1 of the Westinghouse report regarding clad oxidation, please provide the expected clad oxidation values for the 422V+ fuel design operation at KNPP.

13. In Section 2.5 of the Westinghouse report regarding Seismic/LOCA impact on fuel assemblies, the analysis results demonstrated adequate grid load margin for all fuel assemblies except the fuel assemblies on the periphery of the 13 assembly row in the limiting mixed Condition I. Please discuss the various core configurations evaluated as part of this analysis and identify which configuration produces the limiting condition.

14. In Section 2.5.1 of the Westinghouse report, three computer codes are identified for evaluating the fuel assembly and reactor core models. These codes are NKMODE, WECAN and WEGAP. Please provide a reference to the NRC approval for these codes. If not NRC-approved, please provide information regarding control of these codes, particularly with respect to quality assurance, benchmarking, validation and verification. If used to evaluate Framatome ANP fuel, justify their applicability.

15. Section 3.1 of the Westinghouse report states that standard nuclear design analytical models and methods (Westinghouse computer codes and methods) accurately describe the neutronic behavior of the 422V+ fuel design. Please discuss the codes and methods applied to evaluate the current Framatome ANP fuel through the transition cores at KNPP. Provide technical justification for the applicability of these codes and methods.

16. Section 3.1 of the Westinghouse report states that marked-up changes to the TSs are included in Appendix A. Section 3.6 states that the TS changes which impact the nuclear design for 422V+ fuel are modifications to the protection trip setpoints (summarized in Section 5.1). The Westinghouse report does not contain an Appendix A and Section 5.1 does not summarize the protection trip setpoints. Please provide this information.

17. Section 3.2 of the Westinghouse report states that, “The effects of extended burnup on nuclear design parameters have been previously discussed in WCAP-10125-P-A, “Extended Burnup Evaluation of Westinghouse Fuel,” and that discussion is valid for the anticipated 422V+ design discharge burnup.” What is the technical basis for this conclusion? The 422V+ product design is extended to a lead rod burnup of 75,000 MWD/MTU. WCAP-10125-P-A discusses a lead rod average burnup of 60,000 MWD/MTU.

18. Section 3.4 of the Westinghouse report states that four core models were developed and used in the nuclear design analyses performed. However, this section only discussed three of the models; first transition cycle, second transition cycle and a third “all 422V+” core. Please discuss the fourth core design evaluated. Also, were

NRC-approved methods employed in determining the transition core configurations evaluated or is this based on standard reload methodology?

19. Section 3.5 of the Westinghouse report addresses power distributions and peaking factors. The terminology used here is not clear. Westinghouse refers to the factor $F_Q^N(z)$ as the total peaking factor in this section and in Figure 3.8. Typically, the total peaking factor is identified as F_Q^N, and is not a function of height (z). Is Figure 3.8 shown for a particular axial location? Please clarify the terminology used and verify that the total peaking factor limit will be met with the new 422V+ fuel design.

20. With respect to nuclear design, please discuss how the acceptance criteria specified in NUREG-0800, SRP Section 4.3 are ensured to be satisfied for the 422V+ fuel design.

21. Section 4.2 of the Westinghouse report states that the use of the WRB-1 DNBR correlation with a 95/95 correlation limit of 1.17 is applicable for the 14x14 V422+ fuel assemblies. The NRC staff SER for WCAP-8762 "New Westinghouse Correlation WRB-1 For Predicting Critical Heat Flux in Rod Bundles With Mixing Vane Grids," states that the WRB-1 correlation may be used for 15x15 and 17x17 optimized fuel assembly design. The justification provided by Westinghouse for the application of WRB-1 and its associated limit to 14x14 fuel assemblies is based on Westinghouse responses to requests for additional information (RAI's) on WCAP-8691, "Fuel Rod Bow Evaluation." RAI responses do not constitute NRC staff approval. Additionally, it is not apparent where this issue was discussed in the referenced RAI responses. Please provide the technical basis for application of the WRB-1 correlation and correlation limit of 1.17 for the 14x14 V422+ fuel assembly designs. In your response, please discuss grid design differences between the fuel types.

22. The licensee is proposing to revise the nuclear enthalpy rise hot channel factor equation by changing the reduction term from [1+0.2(1-P)] to [1+0.3(1-P)]. Please provide the technical basis for this change.

23. Section 4.2 of the Westinghouse report discusses the hydraulic compatibility of the various fuel assemblies.

    a. Regarding the conclusion that the fuel assembly crossflow is well within the bounding Westinghouse experience base of transition core analysis, please provide quantitative results of the significant parameters of interest which would justify this conclusion. For example, the response to this question may include a table comparing ΔP values for the Framatome ANP fuel, 422V+ fuel and Westinghouse OFA fuel.

    b. Westinghouse states that the Framatome ANP fuel assemblies may experience an increase in assembly lift forces of approximately 10 percent once the Westinghouse 422V+ fuel assemblies are loaded in the core. Please discuss the consequences of this increase and justify why such an increase is acceptable under normal and transient operating conditions.

24. Section 4.6 of the Westinghouse report discusses the transition core effect and the calculation of a transition core DNBR penalty. The calculated penalty is shown in Figure 4-5 and in the equation in Section 4.6. Figure 4-5 shows the relationship between fraction of 422V+ in the core and the DNBR penalty to be linear, and shows results for two transition core configurations of approximately 10 percent and 55 percent 422V+ fuel loaded in the core. Is this linear relationship based on actual analysis or is it an assumption? If this is based on actual date, then please provide a plot including the data points. If this is an assumption, then please justify that the DNBR penalties to be used for the 1st and 2nd transition cycles are conservative.

25. Table 4-4 of the Westinghouse report details the DNBR margins available.

    a. Please show how the DNBR design limit of 1.24 was calculated.

    b. The DNBR safety limit of 1.34 was selected to conservatively bound the effects of rod bow, transition core and any other DNBR penalties that may occur, and to provide operating flexibility. The margin between the DNBR design and safety limits is shown as 7.46 percent. However, the DNBR penalties listed in this table sum to approximately 8.4 percent. Therefore, the retained DNBR margin does not appear to be sufficient. Does the DNBR safety limit need to be further increased to account for these effects?

26. Section 5.1 of the Westinghouse report indicates that a T_{AVG} range of 556.3 °F to 573.0 °F was considered in the Non-LOCA transients reanalyzed for the fuel transition and power uprate program. The KNPP draft COLR requires that during steady state power operation T_{AVG} shall be < 576.7 °F. Assuming a lower temperature in the transient analyses is non-conservative with respect to DNBR. Please discuss this discrepancy, including why DNBR limits are not exceeded during anticipated operational occurrences.

27. Section 5.1 of the Westinghouse report indicates that up to 10 percent symmetric steam generator tube plugging was considered in the analyses. Please provide a discussion of the expected impacts of this level of asymmetric tube plugging on the non-LOCA transient results.

28. Inadvertent loading and operation of a fuel assembly in an improper position is not listed in Table 5.1.7 as an event that was reanalyzed as part of this amendment request. Please discuss the reason for this and why this event remains acceptable under the proposed fuel transition and uprate power program. Is this event not part of the KNPP licensing basis?

29. Tables 5.1-7 and 5.1-1 of the Westinghouse report provide a summary of the methods used and the results of the non-LOCA transient analyses. The information provided appears to be for the Westinghouse fuel only, as only the Westinghouse DNB correlations are listed. Please provide the analogous results which demonstrate the acceptability of the Framatome Fuel over the transition operating cycles.

30. Regarding the reanalysis of the Uncontrolled RCCA Withdrawal from a Subcritical Condition transient:

    a. Please provide the technical basis for the assumptions of an 1100 pcm Doppler power defect, and the change in initial power level from 1.0E-13 to 1.0E-9. Are these conservative assumptions?

    b. Please verify that maximum values of the effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (l*) are used in the analyses.

    c. Table 5.1-1 provides the DNBR results for the W-3 correlation only, which is valid below the first mixing vane grid. Please provide the MDNBR results for above the first mixing vane grid, which are calculated using the WRB-1 correlation. Also, please provide the basis for the analysis limit of 1.39 for the W-3 correlation.

31. Regarding the reanalysis of the Uncontrolled RCCA Withdrawal from Power transient:

    a. RETRAN (a system code) rather than a subchannel code such as VIPRE is used for DNBR analysis for this transient. The use of the RETRAN DNBR model requires certain user-input values (not listed here because this is shown as proprietary on page 55 of WCAP-14882-P-A). Please discuss how this user-input was determined for KNPP for the fuel transition and power uprate program.

    b. NUREG-0800, SRP, Section 15.4.2 lists the acceptance criteria for this event as DNBR>limit and fuel centerline temperature remains less than the melt temperature. Please provide quantitative results which demonstrate that the fuel centerline temperature acceptance criteria is satisfied.

    c. The tables of results are shown for three power levels: 100 percent, 60 percent and 10 percent power. What is the 100 percent power value (MWt) assumed in the analyses for minimum DNBR and fuel centerline temperature? Did the analysis include calorimetric uncertainty?

32. Regarding the reanalysis of the RCCA Misalignment transient:

    a. NUREG-0800, Standard Review Plan, Section 15.4.3 lists the acceptance criteria for this event as DNBR>limit and fuel centerline temperature remains less than the melt temperature. Please provide quantitative results which demonstrate that these acceptance criteria are satisfied for each of the cases analyzed.

    b. Please list and provide a reference to the nuclear physics computer codes used to analyze the steady-state power distributions.

    c. Please provide the maximum value of the resulting radial power peaking factor ($F^N_{\Delta H}$). When in core life does this value occur?

d. What power level (MWt) is assumed in the analyses for minimum DNBR and fuel centerline temperature? Did the analysis include calorimetric uncertainty?

33. Regarding the reanalysis of the Chemical and Volume Control System Malfunction:

a. The maximum dilution flow rate used in the analysis was decreased from 180 gpm to 80 gpm. The KNPP USAR states that 80 gpm is the maximum dilution flow with two charging pumps operating and three letdown orifices in service. Please provide the technical basis for reducing the dilution flow assumption. Is there ever a situation in which all three charging pumps can be operating and contributing to the dilution flow? Demonstrate that the 80 gpm assumption is conservative.

b. The initial boron concentration values have been changed in the analyses. During refueling, the value increased from 2200 to 2250 ppm. During Startup, the value decreased from 2200 to 1800 ppm. During power operation, the value increased from 1600 to 1780 ppm. Please provide the technical basis for these changes.

c. NUREG-0800, SRP, Section 15.4.6 also lists DNBR and primary and secondary system pressure acceptance criteria for this event. Please discuss the analyses performed and provide results which verify that these acceptance criteria are satisfied.

d. Please provide the calculated available operator action times (time between alarm and loss of shutdown margin) for the three modes of operation analyzed. How did the operator action time change from the previous analysis of record?

34. Regarding the reanalysis of the RCCA Ejection event:

a. NUREG-0800, SRP, Section 15.4.8 and Regulatory Guide 1.77 also list reactor pressure as an acceptance criteria for this event. Please discuss the analyses performed and provide results which verify that this acceptance criteria are satisfied.

b. Proposed USAR Section 14.2.6, insert B states that a minimum value of the Doppler power defect and moderator feedback is assumed in the analyses. Does this imply an absolute value, such that a least negative value for these parameters is used?

c. USAR Table 14.2.6-1 shows the maximum cladding average temperature for the EOL-HZP case to be 2987 °F. The analysis limit according to the Westinghouse methodology is 2700 °F. Based on this, please explain why this is acceptable? Is cladding failure assumed in this case?

d. Please discuss how the Doppler feedback reactivity weighting factors are calculated and applied. It would be helpful if this could be demonstrated through a simple sample calculation. Please provide a reference to the NRC-approved methodology for these factors.

35. Please expand Table 5.1.7 of the submittal to include the following information regarding the computer codes and methodologies used in the new analyses in a tabulate form, include ANC and PHOENIX: 1) the computer codes and methodologies used in each of the transient and accident analyses, 2) the staff review and approval status of these codes and methodologies, 3) the conditions and restrictions for each of the code and methodologies, 4) how these conditions and restrictions are satisfied in each application to the new analyses, and 5) is the use of these codes and methodologies in the new analyses consistent with the current licensing basis.

36. Please expand Table 5.1.1 of the submittal to include the following information regarding the assumptions, and results of the new analyses of the new analyses in a tabulate form: 1) initial plant conditions consistent with that operated at uprated power level, 2) other major assumptions used in the new analyses including the assumed limiting single failure and loss of offsite power during the event, 3) state the acceptance criteria applied to each of event analyzed, 4) identify any assumptions and acceptance criteria used in the new analyses that are deviate from the current licensing basis and provide justification for the differences, and 5) provide the results of the new analyses for each event to demonstrate that the consequences of each event analyzed met their acceptance criteria with sufficient margin.

37. Please provide a tablation to compare the plant operational parameters for the current Rated power level and the uprated power level.

38. Provide technical basis to justify that the safety analyses performed at an uprated power level bound the plant operation at the current rated power level. Please address the affect of different initiate conditions to the analyses.

39. Proposed UFSAR Table 14.0-2 reflect that there is a 2 percent calorimetric uncertainties considered in the assumed power level in all the safety analyses not associated with DNB calculation using the Westinghouse Revised Thermal Design Procedure (RTDP). However, this fact is not clearly indicated in other discussions of the submittal. Please confirm that this the new analyses to support the proposed fuel change and future power uprate is based on an assumed core thermal power of 1807.5 MWt.

40. It is indicated in the revised UFSAR that TS 2.1.2, 3.10m and the TS Basis for TS 2.2, 3.4, 3.10k, 3.10.l, 3.10m were changed. Describe the transient and accident analyses to which these changes were affected and explain the need for these changes.

41. TS Basis 2.1-1 indicates that the settings of the power operated relief valves (PORVs), have been established to prevent the primary system exceeding its safe limit of 2735 psig for all transients except the RCCA ejection accident. Identify the transient in which the PORVs are credited for mitigating the overpressure event. Please confirm that the PORVs and their associated controls at KNPP are designed to safety grade standard so that the PORVs are qualified for accident mitigation.

42. Please explain the need and the basis for changes made in TS B 3.4-2 regarding auxiliary feedwater system.

43. Please explain the basis for the increasing percentage of fuel rod experiencing DNB from 40 percent to 50 percent for a locked rotor event. Confirm that the radiological consequences are within the guidelines of 10 CFR Part 100 limits with such large amount of failed fuel.

44. Page 14-0-6 of the revised UFSAR indicated that there are changes to reactor protection setpoints (e.g. overpower delta T, over-temperature delta T, RCS low flow, pressurizer high level, and pressurizer low pressure). Please explain why these changes are not included in the list of proposed TS changes.

45. Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator (SG) water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG water level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, anticipated transient without scram (ATWS) and steamline break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affect the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses . The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how KNPP account for these uncertainties documented in these advisory letters in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still adequate.

46. Please explain why the following events are not included in the transient and accident analyses: 1) feedwater line break accident, 2) steam generator tube rupture accident, and 3) inadvertent startup of emergency core cooling system at power.

47. Provide evaluation of the impacts of the proposed fuel changes and future power uprate on the ability of KNPP to cope with a Station Blackout event.

48. To show that the referenced generically approved Best Estimate LOCA analysis methodologies apply specifically to the Kewaunee plant provide a statement that NMC and its vendor have ongoing processes which assure that LOCA the values and ranges of analysis inputs for peak cladding temperature-sensitive parameters bound the values and ranges of the as-operated plant for those parameters.

49. If Kewaunee is sharing BE large-break loss-of-coolant accident (LBLOCA) analyses with any other plant(s), are the Kewaunee “plant-specific” analyses based on the model and or analyses of any other plant? If, so justify the that the model or analyses apply to Kewaunee. (E.g. if the other design has a different vessel internals design the model wouldn’t apply to Kewaunee.)

50. Is NMC applying the same Kewaunee “plant-specific” analyses to any of the other units? If so, please justify the applicability of the Kewaunee analyses to those plants.

51. For the loss of external electrical load transient at 102 percent power, your revised UFSAR indicates that you use 2200 psia for the initial condition when modeling the pressurizer pressure. Given that you operate at 2250 psia and that your uncertainty is ±50psi, explain how the 2200 psia assumption is more limiting than a 2300 psia for the overpressure condition. Also, if you use 2300 psia as the starting condition, what peak pressure is achieved? Is this pressure below 110 percent of your design values?

52. Discuss the basis for assuming an initial pressurizer pressure of 2250 for safety analyses concerning DNB. Why is this assumption resulting a most conservative minimum departure from nucleate boiling (MDNB) value?

53. Discuss the basis for assuming a reactor vessel coolant flow of 186,000 gpm as the most conservative conditions in calculating MDNB value.

54. With respect to the proposed uprated power level, please demonstrate that your LOCA analysis models and your emergency procedures are consistent with 10 CFR 50.46 requirements (1) during the switch-over from the refueling water storate tank to the containment emergency sump and (2) during long term cooling while the emergency core cooling system (ECCS) pumps draw water from the containment emergency sump. Note the general intention of this request is to obtain confirmation that ECCS pump operation, flow rates, and water injection locations will be consistent with 10 CFR 50.46 requirements.

55. With respect to the uprated power generation rate, please describe the containment emergency sump configuration, including the screen design, and show that an adequate water source is provided to the ECCS during operation after initiation of recirculation from the emergency sump. Your response should include consideration of the effect of potential debris accumulation on the effective water pressure and flow rate available to the ECCS pump suctions.

56. Has the NRC approved the Westinghouse Best Estimate LBLOCA methodology for analyses of Framatone fuel in any other 2-loop upper plenum injection plant?

57. Discuss how the Kewaunee LBLOCA methodology(ies) account for the mixed fuel types.

58. Because it is possible for one fuel type to be PCT-limiting, and another to be oxidation-limiting, provide a commitment to report LOCA analysis results for all fuel types (represented in a significant number of assemblies), per 10 CFR 50.46(a)(3).

WCAP-15591, Rev. 0, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis and 54F Replacement Steam Generators)."

1. Provide a plant-specific basis for the parameter uncertainties and sensitivities listed in Tables 1 through 9 of WCAP-15591, including the source of each value and/or the method by which each value was determined.

2. Kewaunee Assessment Process (KAP) No. 00-2991: Update updated safety analysis report (USAR) to Reflect Actual reactor coolant system (RCS) Bypass Flow Loop Indication and Alarm [UCR #R16-046 - Page 7.2-34] (ADAMS Accession No. ML003775364) stated, “This KAP was initiated to evaluate the discrepancy between the USAR and the actual reactor coolant system (RCS) bypass flow loop indication and alarms. The USAR [Updated Safety Analysis Report] described a common low flow alarm with individual status lights for each RCS bypass loop. The actual configuration has individual loop alarms and computer inputs, with no status lights. The function of the alarms/indication, to provide immediate indication of a low flow condition in the bypass loops, was met by the current plant configuration. . . .”

   In the discussion on page 9 of WCAP-15591, Westinghouse stated, “Allowances are made (as noted on Table 2) for hot leg and cold leg streaming, resistance temperature detector (RTDs), turbine pressure transmitter, process racks/indicators and controller. Based on one T_H and one T_C RTD per channel to calculate T_{avg} and with the RTDs located in the hot and cold leg bypass manifolds, the calculated electronics uncertainty using Equation 3 is [ ]$^{+a,c}$.”

   Provide a technical basis for including hot leg temperature streaming in the uncertainty calculation.

3. In the discussion on page 9 of WCAP-15591, Westinghouse stated, “A bias of [ ]$^{+a,c}$ for T_{cold} streaming (in terms of T_{avg}) based on a conservative [ ]$^{+a,c}$ T_{cold} streaming uncertainty is included in Table 2. . . .”

   Provide the technical basis for the cold leg temperature streaming bias, including a discussion of cold leg temperature streaming effects in Kewaunee Nuclear Power Plant.

4. On page 14 of WCAP-15591, Westinghouse states, “The thermal output of each steam generator is determined by a secondary side calorimetric measurement that is defined as: . . . . The steam enthalpy is based on the measurement of steam generator outlet steam pressure assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and a calculated feedwater pressure.” A similar statement regarding the calculation of feedwater pressure was made on page 26 in the discussion describing the calculation of reactor power.

   On page 15, Westinghouse states, “The calorimetric RCS flow measurement is thus based on the following plant measurements: . . . Feedwater pressure (P_f) . . . .” Footnote (2) in Table 3 states that feedwater pressure is measured with a transmitter (PT-21196) and digital acquisition system on the feedwater bypass loop.

Explain why a calculated feedwater pressure value is used in the secondary side calorimetric calculation, and a measured feedwater pressure value is used in the calorimetric RCS flow measurement uncertainty calculation and in the calculation of reactor power.

5. Provide a derivation of the equation for the controller deadband variance $(s_2)^2$ that is provided on page 9 of WCAP-15591.

6. On page 16 of WCAP-15591, Westinghouse states, “For the measurement of feedwater flow, the feedwater venturi is calibrated by the vender in a hydraulics laboratory under controlled conditions to an accuracy of [$]^{+a,c}$. The calibration data that substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [$]^{+a,c}$ is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [$]^{+a,c}$ . . . .” A similar discussion is presented on page 26.

   Provide a discussion of the basis for the additional uncertainty value that accounts for installation effects.

7. On page 16 of WCAP-15591, Westinghouse states, “The uncertainty applied to the feedwater venturi thermal expansion correction (F_a) is based on the uncertainties of the measured feedwater temperature and the coefficient of thermal expansion for the venturi material, usually 304 stainless steel. For this material, a change of ±1°F in the nominal feedwater temperature range changes F_a by [$]^{+a,c}$ and the steam generator thermal output by the same amount.

   “Based on data introduced into the American Society of Mechanical Engineers (ASME) Code, the uncertainty in F_a for 304 stainless steel is ±5%. This results in an additional uncertainty of [$]^{+a,c}$ in feedwater flow. Westinghouse uses the conservative value of [$]^{+a,c}$.” A similar discussion is presented on page 27.

   Describe the process by which the ASME Code uncertainty in F_a for 304 stainless steel is translated into the stated additional uncertainty in feedwater flow.

8. On page 16 of WCAP-15591, Westinghouse states, “Feedwater venturi ΔP uncertainties are converted to percent feedwater flow using the following conversion factor:

   percent flow = (ΔP uncertainty)(1/2)(transmitter span/90)^2"

   On page 23 of WCAP-15591, Westinghouse states, “. . .The ΔP transmitter uncertainties are converted to percent flow using the following conversion factor:

   percent flow = (ΔP uncertainty)(1/2)($Flow_{max}/Flow_{nominal}$)^2

   where $Flow_{max}$ is the maximum value of the loop RCS flow channel.”

On page 27 of WCAP-15591, Westinghouse states, “Feedwater venturi ΔP uncertainties are converted to percent flow using the following conversion factor:

$$\text{percent flow} = (\Delta P \text{ uncertainty})(1/2)(\text{Flow}_{max}/\text{Flow}_{nominal})^2$$

The feedwater flow transmitter span (Flow_{max}) is 117.5% of nominal flow.”

Provide a derivation of the conversion equation, and provide a basis for each term in the parentheses.

9. On page 17 of WCAP-15591, Westinghouse states, “The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]$^{+a,c}$ of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]$^{+a,c}$ of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2, and by input power measurements from several other plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is [ ]$^{+a,c}$ of core power.”

   Provide a technical basis for each of the estimated uncertainties, and the process by which these uncertainties were combined and then subsequently used to determine the uncertainty value for core power.

10. On page 17 of WCAP-15591, Westinghouse states, “. . . The uncertainties for the instrumentation are noted in Table 3 and the sensitivities are provided on Table 4. The hot leg streaming is split into random and systematic components. For Kewaunee where the RTDs are located in RTD bypass manifolds, the hot leg temperature streaming uncertainty components are [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ systematic.”

    Provide the basis for the hot leg temperature streaming uncertainty component values.

11. On page 17 of WCAP-15591, Westinghouse used the values from Table 5 in the 2-loop uncertainty equation (with biases) to calculate the flow uncertainty. The Table 5 bias values appear to be inconsistent with the resulting stated flow uncertainty bias value.

    Provide the calculation that results in the flow uncertainty bias value provided on page 18.

12. On page 23 of WCAP-15591, Westinghouse states, ". . . The loop RCS flow uncertainty is then combined with the calorimetric RCS flow measurement uncertainty. This combination of uncertainties results in the following total flow uncertainty:

# of loops	flow uncertainty (% flow)
2	±2.86 (random)
	±0.11 (bias)

The corresponding value used in RTDP is:

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

A similar discussion regarding feedwater venturi ΔP uncertainties is presented on page 27.

Provide the calculation details that resulted in the percent flow uncertainties and the percent flow standard deviation.

<u>LIST OF MEETING ATTENDEES ON JANUARY 15, 2003</u>

Name	Company
Tom Webb	NMC
Frank Orr	NRC
Mike Waterman	NRC
Francis Akstulewicz	NRC
Fred Baskerville	Westinghouse
Mark Kowal	NRC
Chu-Yu Liang	NRC
John G. Lamb	NRC
John T. Holly	NMC
Jeffrey L. Voskuil	NMC
Jerry Riste	NMC
William Slagle	Westinghouse
Azad Meliksetian	Westinghouse
Roy H. Owoc	Westinghouse
Scott Snider	Westinghouse
Walter Mooman	Westinghouse
Shih-Liang Wu	NRC
Anthony Attard	NRC
L. Raghavan	NRC

ENCLOSURE 2

Kewaunee Nuclear Power Plant

cc:

John H. O'Neill, Jr.
Shaw Pittman, Potts & Trowbridge
2300 N. Street, NW
Washington, DC 20037-1128

David Zellner
Town Chairman - Town of Carlton
N2164 County B
Kewaunee, WI 54216

Larry L. Weyers
Chairman, President, and CEO
Wisconsin Public Service Corporation
600 North Adams Street
Green Bay, WI 54307-9002

Resident Inspectors Office
U.S. Nuclear Regulatory Commission
N490 Hwy 42
Kewaunee, WI 54216-9510

Regional Administrator
Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Ave M. Bie, Chairperson
Public Service Commission of Wisconsin
PO Box 7854
Madison, WI 53707-7854

Roy Anderson
Chief Nuclear Officer
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

David Molzahn
Nuclear Asset Manager
600 North Adams Street
Green Bay, WI 54307-9002

Sarah Jenkins
Electric Division
Public Service Commission of Wisconsin
PO Box 7854
Madison, WI 53707-7854

NRC NRR Project Manager
Mail Stop O-8-H-4A
U. S. Nuclear Regulatory Commission
Washington, DC 20555