NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

January 30, 2004

NRC-04-012
10 CFR 50.90

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 NRC Clarification Questions On Responses To Requests For Additional Information Regarding License Amendment Request 195, Stretch Power Uprate For Kewaunee Nuclear Power Plant (TAC NO. MB9031)

References:
1) Letter NRC-03-057 from Thomas Coutu to Document Control Desk, "License Amendment Request 195, Application for Stretch Power Uprate for Kewaunee Nuclear Power Plant," dated May 22, 2003.
2) Letter from Thomas Coutu to Document Control Desk, "Responses to Requests for Additional Information and Supplemental Information Regarding License Amendment Request 195, Stretch Power Uprate For Kewaunee Nuclear Power Plant," dated November 5, 2003.
3) Letter from Thomas Coutu to Document Control Desk, "Responses To NRC Clarification Questions On Responses To Requests For Additional Information Regarding License Amendment Request 195, Stretch Power Uprate For Kewaunee Nuclear Power Plant," dated December 15, 2003.
4) Letter from John Lamb (NRC) to Thomas Coutu (NMC), "Kewaunee Nuclear Power Plant - Review Of License Amendment Request No. 195, Stretch Power Uprate (TAC NO. MB9031)," dated January 26, 2004

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) submitted license amendment request (LAR) 195 (Reference 1) for a stretch power uprate of six percent. The stretch 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 1673 MWt to 1772 MWt.

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On November 5, 2003 and December 15, 2003, NMC responded to requests for additional information from the Nuclear Regulatory Commission (NRC) regarding the proposed stretch power uprate (References 2 and 3). Subsequent to NMC's response, the NRC staff has requested clarification on some of the responses submitted in NMC's November 5th and December 15th letters. This letter and enclosure contains the NMC responses to the NRC requests for clarification.

Enclosure 1 contains the clarification questions from the NRC. These questions were derived from emails and conference calls with the NRC staff. Enclosure 2 contains NMC's response to these clarification questions. Enclosure 3 is Main Steam piping velocities. Enclosure 4 contains the list of regulatory commitments with updated due date status and one additional commitment. Enclosure 5 contains NMC's response to the NRC staffs' questions contained in reference 4.

These responses do not change the Operating License or Technical Specifications for the KNPP, nor do they change any of the proposed changes to the Operating License or Technical Specifications in reference 1. The responses do not change the no significant hazards determination, the environmental considerations, the requested approval date, or the requested implementation period originally submitted in reference 1.

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

Summary of Commitments

This letter makes the following new commitments:

1) NMC will provide NRC with a status update and schedule for resolution of GL 96-06 water hammer issues by April 2, 2004.

If there are any questions or concerns associated with this response, contact Mr. Gerald Riste at (920)388-8424

I declare under penalty of perjury that the foregoing is true and correct.
Executed on January 30, 2004.

Thomas Coutu

Thomas Coutu
Site Vice-President, Kewaunee Plant
Nuclear Management Company, LLC

GOR

Enclosures: (5)

cc:- Administrator, Region III, USNRC
Project Manager, Kewaunee Nuclear Power Plant, USNRC
Senior Resident Inspector, Kewaunee Nuclear Power Plant, USNRC
Electric Division, PSCW

ENCLOSURE 1

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

January 30, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195

NRC Clarification Questions on Responses to Requests for Additional Information

4 Pages Follow

NRC EMEB Branch Questions

1. Clarify the pipe stress Table 1 in RAI response #55 - Explain why the "Loading Condition" for the Condensate System is "Sustained + Thermal" and the loading condition for Feedwater, Bleed Steam, and Heater Drains is only "Thermal."

2. For RAI question 57 on pump and valve programs, provide more details on how NMC evaluated that the valves were acceptable for uprated conditions, giving examples of typical evaluations for different pump or valve types (e.g., relief valves, MSIVs). Also briefly indicate; 1) why there are no relief valve setting changes or pump change-outs, 2) whether the MSIV has a minimum close time limit, if the increased steam flow could reduce the MSIV closure time, and why the design and operating experience with the MSIV supports a finding that the closure time will not be affected, 3) whether the ambient temperature increase has been evaluated for the GL 95-07 pressure locking and thermal binding issue, and 4) whether Regulatory Commitment #16 on MOV temperature effects had been completed

3. Based on the environmental qualification (EQ) calculations submitted in NMC's December 15, 2003 letter, what is the licensee's overall conclusion concerning the results.

4. RAI question 13 questioned the acceptability of the overdutied circuit breakers on KNPP Buses 1-1 through 1-4. Enclosure J of the November 5 2003 was an evaluation performed on this condition. The NRC staff request information associated with any prior NRC review of this condition and impact of the power uprate on the evaluations conclusions.

5. RAI questions 10 and 16 are associated with the operation of the feedwater and condensate pumps under reduced voltage conditions and into the motors service factor rating. In the November 5, 2003 and December 15, 2003 responses it appears to the staff that the IEEE 666 standard and NEMA MG-1 are being inappropriately combined. The staff requested additional information for supporting evidence that the overloading of these motors based on their service factors wouldn't affect the operation of these motors. The licensee responded to the NRC staff's RAI but the staff still requires further clarification. Describe for the staff the affect of the power uprate (e.g. reduced voltage and use of the pumps service factor) on the probability of a plant trip or transient.

NRC SPLB Branch Questions

1. For BOP systems, the evaluations in Section 4 of the Licensing Report (BOP Interface by Westinghouse) and Section 8 (BOP system evaluations by SWEC) do not have comprehensive, integrated conclusions.

   A. Provide specific conclusions for BOP systems (focus on MS, Condensate, FW, AFW, and TG) addressing the impact of uprated flows, pressures, temperatures, etc.

   B. Why does the Westinghouse design criteria for MS safety valves having 100% normal steam flow capacity not have to be met?

   C. Justify how we can operate the FW heaters beyond their design rating and summarize the inspection results.

D. Explain why the bolting changes were required for the HP and LP turbines and conclude why the upgraded bolts are acceptable.

E. Explain the regulatory basis of the FAC program.

F. Explain the basis of the FW control valve Cv.

2. The Licensee's response to RAI question 4 (November 5, 2003 Letter) stated that the requirements associated with borating the RCS to cold shutdown, utilizing the RWST, with and without letdown in service is being evaluated for any required procedure changes. This evaluation, and update of any required procedures, will be completed prior to implementation of the SUR. This is considered part of our Regulatory Commitment #12. The NRC staff needs to know the results of this evaluation before it can issue a safety evaluation on the acceptability of this amendment request. Please submit a summary of the results of this evaluation.

3. The NRC staff has some additional questions concerning the stretch power uprate.

A. From WCAP 16040-P, page 8-7 "Most calculated MS flow velocities at power uprate do not exceed the previously accepted recommended maximum values for continuous operation." What are calculated and acceptable values?

B. How did/will the licensee ensure new Feed Reg Valve (FRV) Trim is acceptable?

C. How does the licensee ensure sufficient NPSH for the Feedwater pumps? How much margin does the licensee expect in pump suction pressure?

D. Is the Feedwater Isolation Valve (FWIV) acceptable for Stretch Uprate containment response?

E. How is the KNPP Flooding Analysis impacted by the power uprate?

F. Describe the affect of the power uprate on the KNPP missile analysis.

G. Describe the affect of the power uprate on the Turbine overspeed trip settings (reference WCAP 16040-P, p 8-12).

H. From WCAP 16040-P, the licensee stated, "The exhaust end loading at the uprated conditions increases. At this loading the exhaust flow guide bolting may be affected. If desired, upgraded bolting can be installed." What actions did the licensee do regarding turbine exhaust flow guide bolting?

4. In the aftermath of the Three Mile Island, Unit 2, accident the NRC requested, as a part of NUREG 0737, an evaluation of the auxiliary feedwater system. Please describe the basis for the present condensate storage tank level found in KNPP technical specifications and describe how KNPP conforms to the Standard Review Plan, section 10.4.9, branch technical position ASB10-1, item B.1

5. Describe the results of the impact of the power uprate on the GL 96-06 results

NRC SPSB Branch Questions

1. Please describe the assumptions made regarding the behavior of drops formed by the break flow in high energy line break calculations done for structural analysis (Section 8.5

of WCAP 16040-P) and equipment qualification calculations (Section 8.9 of WCAP 16040-P).

2. What single failure assumptions are made in the analyses to support TS 3.3.c.1.A.3?

3. Please verify that the assumption that the break mitigation hardware reduces the break discharge to 50% of the no load flow in the cubicle pressurization analyses is consistent with the current Kewaunee licensing basis. If it is not, please provide a justification for this assumption

NRC SRXB Branch Questions

1. In the November 5, 2003 response, number 34, NMC stated that calculations demonstrate that even if upper plenum injection is established as late as 18 hours after the LOCA (SBLOCA), the vessel boron concentration will still be 4 weight % under the boron precipitation point, assuming a saturation pressure of 35 psia. Other nuclear plants have stated that they will reach the boron saturation point in approximately 6-9 hours (Adams Accession NO. ML032720538). Please discuss this inconsistency and any actions necessary to ensure boric acid precipitation is not a concern.

2. The NRC requests additional information concerning the elimination of the DNBR rod bow penalty for application to KNPP Cycle 26. Specifically, the NRC requests detailed information be provided about the way the FDH burndown was calculated during the cycle and the way the corresponding DNBR credit was calculated.

NRC SPSB-C Branch Question

1. With regard to the waste gas decay tank rupture and volume control tank rupture analyses, you state that in absence of guidance in RG 1.183, you have assumed that the offsite dose acceptance criterion is the same as given for the fuel handling accident (i.e., 6.3 rem TEDE). However, in the staff's safety evaluation (issued March 17, 2003) for implementation of the alternative source term, this acceptance criterion was not used by the NRC staff in its finding. Kewaunee was originally licensed with a site boundary dose acceptance criterion of 500 mrem whole body for these accidents. Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas Tank System Leak or Failure," which is located in SRP Section 11.3, "Gaseous Waste Management Systems," also presents a dose acceptance criterion of 500 mrem whole body. The curie content of these tanks is controlled so that 10 CFR 20.1301 dose limits to individual members of the public are not exceeded in the event of a release.

   A. Please explain how your assumed acceptance criterion of 6.3 rem TEDE was chosen.
   B. Please explain how your assumed dose acceptance criterion of 6.3 rem TEDE for these accidents is consistent with Kewaunee's original licensing and design basis, and Branch Technical Position ETSB 11-5, or why the difference should be found acceptable.

C. Please explain how the assumed accident dose acceptance criterion interacts with control of radioactivity in the tanks and the 10 CFR 20 dose limits for individual members of the public.

NRC EEIB Branch Question:

1. NMC's December 15, 2003 response to NRC item 9 described the setpoint methodology used for the KNPP. The NRC staff requires additional information associated with how channel operability is determined

ENCLOSURE 2

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

January 30, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

NMC Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195

NMC Responses to NRC Clarification Questions

26 Pages Follow

NRC EMEB Questions

1. Clarify the pipe stress Table 1 in RAI response #55 - Explain why the "Loading Condition" for the Condensate System is "Sustained + Thermal" and the loading condition for Feedwater, Bleed Steam, and Heater Drains is only "Thermal."

Response:

Per USAS B31.1.0-1967 piping code, applicable to KNPP BOP piping systems, the following applies to the allowable stress for thermal expansion (Sa):

$$Sa = f\,(1.25\,Sc + 0.25\,Sh) \quad \text{equation (1) from code section 102.3.2}$$

or, when considering the sustained stress margin per Section 102.3.2(d)

$$Sa = f\,(1.25\,Sc + 0.25\,Sh + Sh - Ssus) \;=\; 1.25(Sc + Sh) - Ssus$$

Where: Sc = material allowable stress at minimum cold temperature
Sh = material allowable stress at maximum hot temperature
f = stress range reduction factor (equal to 1.0 in this case)
Ssus = sum of the calculated sustained stresses

For the calculated thermal expansion stress (SE) to be acceptable, the following condition has to be met:

$SE < Sa$, where Sa is determined with or without the sustained stress margin

When considering the sustained stress margin, the following applies:

$$SE < 1.25(Sc + Sh) - Ssus$$

or shown differently,

$$SE + Ssus < 1.25\,(Sc + Sh)$$

For the Feedwater, Bleed Steam, and Heater Drains piping systems, the SE stress levels were all less than Sa (without need for consideration of the sustained stress margin), and were therefore acceptable under the Thermal Loading Condition alone. For the condensate piping, the calculated thermal expansion stress was slightly above the Sa value without including the sustained stress margin, but was less than the Sa value, and therefore acceptable, when the sustained stress margin was included. Hence, for condensate, the condition showing compliance to the Sustained + Thermal Loading Condition (SE + Ssus) was shown in Table 1.

2. For RAI question 57 on pump and valve programs, provide more details on how NMC evaluated that the valves were acceptable for uprated conditions, giving examples of typical evaluations for different pump or valve types (e.g., relief valves, MSIVs). Also briefly indicate; 1) why there are no relief valve setting changes or pump change-outs, 2) whether the MSIV has a minimum close time limit, if the increased steam flow could reduce the MSIV closure time, and why the design and operating experience with the MSIV supports a finding that the closure time will not be affected, 3) whether the ambient temperature increase has been evaluated for the GL 95-07 pressure locking and thermal binding issue, and 4) whether Regulatory Commitment #16 on MOV temperature effects had been completed

Response:

As indicated previously, the KNPP motor operated valve (MOV) and air operated valve (AOV) Programs assume the maximum system design pressures, temperature, and expected valve differential pressures. The maximum expected valve differential pressure includes the momentum pressure associated with the maximum system flow capability.

In the current programs, valve differential pressure (ΔP) is based on either the system safety valve setting or the total shutoff head of the pumps in the system and includes the momentum pressure of the flow through the valve. The system flow capacity is based on the highest flow capable through the system valves or the capacity of the pumps in the system.

The 6% power uprate makes no changes to the settings of any system relief valves nor does it require change out of any of the pumps within any of the systems. Changes to relief valve settings are not required since the post uprate system operating pressures remain the same or are slightly less than the pre-uprate system operating pressures. Pump changes are not required since the required increase in system flows remain within the original pump design capability. The only valves or piping configuration changes required to be changed as part of the uprate was the main feedwater regulating valves. The trim was changed to increase the Cv to allow passing the required uprate flow with a reduced delta P. Although the flow capability of these valves has increased, the design flow of the feedwater system is based on the flow capacity of the main feedwater pumps which have not been changed.

The MOV Program also takes into consideration any changes in ambient conditions resulting from a high energy line break (HELB) incident. Increases in ambient conditions tend to reduce the available MOV motor torque. The Program MOVs inside containment assume the containment design temperature and pressure which was shown to bound the uprate post accident conditions. The valves outside containment have been reviewed for changes in post uprate HELB temperatures. The review indicates the MOV motor torque will remain adequate following a post uprate HELB incident, however, certain calculations need to be revised. These revisions are currently in progress and as indicated in LAR 195 commitment 16, will be completed prior to implementation of the power uprate (see LAR 195 Commitment 16).

Some specific examples of AOVs evaluated are:

- SD-3A/B, Main Steam PORVs - These valves are evaluated at hot shutdown conditions (1005 psig - 547°F) which do not change due to power uprate. These valves are evaluated assuming a ΔP of 1050 psid at rated flow.

- AFW-2A/B, Motor Operated Aux Feedwater pump discharge valves – These valves are evaluated based on a maximum ΔP of 1200 psid with an auxiliary feedwater pump discharge pressure of 1700 psi at rated pump flow less recirculation flow. These conditions do not change as a result of the power uprate.
- RB-V1 – V4, Reactor Building Vent Valves – These valves are evaluated based on the maximum expected system flow at shutdown conditions with a maximum expected ΔP of approximately 3.9 psid. These shutdown conditions do not change as a result of the power uprate.
- MS-1A/B, MSIVs – These valves are reverse acting check valves and are evaluated for a maximum ΔP of 1138 psid, based on the main steam safety valve settings, which are not changing for uprate. The calculated increased ΔP for the MSIVs associated with the uprate steam flow is approximately 0.4 psid. This small increase in ΔP and the increased uprate steam flow will tend to enhance the closing of these valves and ensure the required maximum 5 second[1] closing time is met at uprate conditions. There is no minimum closing time for the MSIVs. The valves are designed to close under steam line break accident flow conditions that are not impacted by the 6% power uprate.

Some specific examples of MOVs evaluated are:

- MS-100A/B, Steam Generator A/B steam to the turbine driven Aux Feedwater pump isolation– These valves along with the MS-102, Turbine driven Aux Feedwater pump main steam isolation, are evaluated based on a maximum ΔP of 1085 psid with flow conditions associated with a faulted steam generator, main steam line rupture, and a steam generator tube rupture. The 1085 psid ΔP corresponds to the setpoint of the lowest steam generator safety valve, and the flows are limited by the flow capability of the installed valves, neither of which change due to power uprate.
- AFW 10A/B, Aux Feedwater Pump discharge crossover valves – These valves are evaluated based on a maximum ΔP of 1451 and 1453 respectively, based on normal no load steam pressure of 1005 psig. None of the conditions associated with the valve ΔP change as a result of the power uprate.
- FW-12A/B, Steam Generator A/B feedwater isolation valves - These valves are evaluated based on the maximum ΔP based on the discharge pressure of the condensate pumps operating with a recirculation flow of 2200 gpm. The power uprate required no changes to the condensate pumps or the recirculation flow rate, therefore the power uprate does not impact the operation of these valves.

As indicated in our previous response, issues such as the valve thermal binding and pressure locking (GL 95-07) and thermally induced over pressurization of water filled piping inside containment (GL 96-06) were determined not to be impacted, because the new post uprate conditions are bounded by existing program boundary conditions. Both programs assume containment design basis pressure and temperature values for post accident conditions. The uprate containment post accident conditions remain bounded by the design temperature and pressure values therefore, these programs remain unaffected.

[1] KNPP USAR, *revision 18, section 14.2.5*

3. Based on the environmental qualification (EQ) calculations submitted in NMC's December 15, 2003 letter, what is the licensee's overall conclusion concerning the results.

Response:

All environmentally qualified (EQ) electrical equipment at KNPP has been evaluated for the resulting environmental conditions resulting from the 6% stretch power uprate and will remain environmentally qualified.

4. RAI question 13 questioned the acceptability of the overdutied circuit breakers on KNPP Buses 1-1 through 1-4. Enclosure J of the November 5, 2003 was an evaluation performed on this condition. The NRC staff request information associated with any prior NRC review of this condition and impact of the power uprate on the evaluations conclusions.

Response:

WCAP-16040-P NSSS/BOP Licensing Report discusses this on page 8-74, under the heading "4160 VAC Switchgear", it states "There were no new equipment installed or modifications to equipment that changed the short circuit characteristics of the installed equipment at the medium voltage level." This statement means that there is no change in the available three phase or asymmetrical fault current at any plant 4160 Volt bus due to the uprate. The same page says that the bus 1-1 through 1-4 overduty condition "...was previously evaluated and found acceptable." This statement refers to an NRC letter to WPSC dated Jun 2, 1992; see 3.1.1 on page 3, titled "Fault Duty on 4.16kV System".

5. RAI questions 10 and 16 are associated with the operation of the feedwater and condensate pumps under reduced voltage conditions and into the motors service factor rating. In the November 5, 2003 and December 15, 2003 responses it appears to the staff that the IEEE 666 standard and NEMA MG-1 are being inappropriately combined. The staff requested additional information for supporting evidence that the overloading of these motors based on their service factors wouldn't affect the operation of these motors. The licensee responded to the NRC staff's RAI but the staff still requires further clarification. Describe for the staff the affect of the power uprate (e.g. reduced voltage and use of the pumps service factor) on the probability of a plant trip or transient.

Response:

The concern here falls under the category of a Condition II event. The KNPP USAR definition (pg. 14.0-2) is "Condition II occurrences include incidents, any one of which may occur during a calendar year for a particular plant." Feedwater System Malfunction is listed as a Condition II incident. Failure of a Main Feedwater or Condensate Pump motor could therefore cause a Condition II event.

KNPP has operated for 29 years. In that time there has been one Main Feedwater Pump motor failure and one Condensate Pump motor failure.

As you stated in RAI #10, the life of the [motor] insulation is reduced by 50% if the temperature is increased by 8-12 degrees C [above motor nameplate full load temperature]. As noted in our previous response, the condensate pump motor temperature is expected to increase by about 2.3 degrees C and the feedwater pump motor by about 3.9 degrees C due to uprate. Conservatively assuming these motors are operating at full load temperature continuously prior to uprate, the impact of the added 4 degrees C should be less than a 25% reduction in motor insulation life. Therefore the motors should operate for 29 years x 0.75 = 21.75 years between failure (at worst) after uprate. The motor failure is much less frequent than once a calendar year, therefore such failures would remain Condition II incidents and there would not be a more than minimal increase in the frequency of the Feedwater System Malfunction.

A Probabilistic Risk Assessment (PRA) evaluation was performed on the effect of a 25% increase in the feedwater pump motor failure rate, which showed that the change in KNPP Core Damage Frequency (CDF) was less than the 1.0E-6 Delta-CDF criteria for a significant risk impact. In addition, there was no change in the calculated Large Early Release Frequency (LERF) due to this increase in motor failure rate.

NRC SPLB Branch Questions

1. For BOP systems, the evaluations in Section 4 of the Licensing Report (BOP Interface by Westinghouse) and Section 8 (BOP system evaluations by SWEC) do not have comprehensive, integrated conclusions.

   A. Provide specific conclusions for BOP systems (focus on MS, Condensate, FW, AFW, and TG) addressing the impact of uprated flows, pressures, temperatures, etc.

Response:

Main Steam (MS) – The Main Steam system was evaluated for expected power uprate pressure, temperature, and flow velocity conditions. Power uprate will result in a Main Steam system flow rate increase of approximately 8.4 percent with a steam generator steam outlet operating pressure increase of approximately 7.6 percent. The MS system design pressure of 1100 psia bounds the power uprate operating condition of 797 psia. Additionally, the highest normal operating pressures and temperatures occur at no load conditions, which are not affected by the power uprate. The Main Steam pressures and temperatures associated with power uprate are bounded by the designs of the MS system and its components with margin, and are therefore acceptable.

Most calculated Main Steam system flow velocities at power uprate do not exceed the previously accepted recommended maximum values for continuous operation. The exception is that the calculated flow velocities through the 18- and 8-inch diameter lines to the steam dump valves exceed the recommended maximum value (Stone & Webster standard practice) at current and power uprate conditions. However, since the Steam Dump Systems are operated in an intermittent manner and at conditions less than full power (that is, during plant start up and cooldown, and sudden load reductions), which will not change with the implementation of power uprate, the piping velocities are acceptable for the power uprate. Therefore, the Main Steam system flow velocities at power uprate remain acceptable.

The Main Steam (MS) System is included in KNPP's FAC program. The FAC Program owner has a corrective action item to update the FAC program for all impacted systems, including Main Steam, for the stretch uprate (see LAR 195, Regulatory Commitment Number 2).

Condensate and Feedwater (CD & FW) – A hydraulic flow model of the Condensate and Feedwater Systems was used to evaluate system performance under normal and transient conditions. The model analyses included runs to simulate existing plant conditions, and to evaluate plant power uprate normal and transient conditions. The expected system pressures and temperatures were obtained from the power uprate heat balances and the hydraulic model calculation. These values were compared to the design specifications. The system design pressure envelops the system pressures at power uprate, both during normal operating and transient conditions.

With the exception of the piping from feedwater heater 14 to the FW pumps suction, all piping for power uprate is bounded by the design temperature. For this piping section, power uprate conditions show a temperature of 366°F, 16°F above the design temperature of

350°F. The KNPP Piping Specification, however, allows a maximum temperature of 650°F for the same pipe at a maximum pressure of 400 psi. As this pressure is not exceeded at power uprate conditions, this section of pipe is considered acceptable. This design discrepancy was resolved and documented within the KNPP corrective action program.

The valve trim in the feedwater regulating valves (FRVs) was determined to be undersized to support the required normal and abnormal operating requirements at uprated conditions. During the R26 refueling outage in April 2003 the trim in the FRVs was changed out as recommended by the uprate evaluation.

The new operating points for the condensate pumps are within design parameters for flow, head, and NPSH. Therefore, the condensate pumps have sufficient head and capacity margin to support the power uprated plant conditions.

The new operating point for the FW pumps is within the design curve limits for flow, head, and NPSH. Therefore, the FW pumps have sufficient head and capacity margin to support the power uprated plant conditions.

The feedwater heater analysis is discussed in a separate question below.

Auxiliary Feedwater (AFW). – No AFW performance changes (pressure, temperature, or flow) are required as a result of the power uprate. Therefore, acceptable AFW flow performance has been demonstrated for the power uprate conditions, since the existing AFW performance does not change.

Higher uprate decay heat during the four hour Station Blackout coping period necessitated an increase in Condensate Storage Tank (CST) usable volume from 39,000 gallons to 41,500 gallons. Therefore, with the required change in the CST inventory, the AFW system is acceptable for uprate. This inventory change was included in the requested change to Tech Spec 3.4.c for the stretch uprate.

Turbine Generator – Siemens Westinghouse, turbine generator original equipment manufacture (OEM), performed the evaluations of the turbine and generator to support a nuclear steam supply system (NSSS) power of 1780 MWt (1772 MWt + 8 MWt reactor coolant pump (RCP) heat). An evaluation of the impact of the 7.4% uprate conditions on the existing turbine missile analysis and torsional analyses showed the existing missile and torsional analyses remain bounding at uprated conditions.

The turbine evaluations identified that the volumetric steam flow for a 7.4% power uprate with a Tave of 562°F and predicted steam generator pressure (~715 psig) could not be accommodated by the existing high pressure (HP) turbine nozzle block without modification. It was determined that increasing Tave to 572°F would reduce the HP turbine steam volumetric flow requirement to within the capacity of the existing nozzle block and also retain a minimal margin to a valves-wide-open (VWO) condition.

The turbine evaluations contained the OEM recommendation to reduce the maximum allowable mechanical overspeed settings from 111% to 109% and the electronic redundant overspeed trip (ROST and AEH) from 111.5% to 109.5% to ensure the turbine generator design overspeed of 120% is not exceeded at the 7.4% uprated power. In addition, the

Load Drop Anticipator (LDA) settings would also need to be appropriately revised for the 7.4% uprate conditions.

During the R26 refueling outage in April 2003, the necessary changes to the turbine overspeed instrumentation were implemented. The current mechanical overspeed setting is within this specified range and did not require adjustment.

The OEM recommendations also included changing out selected HP Turbine horizontal flange bolts and low pressure (LP) Turbine coupling bolts to a higher strength material to minimize the potential of a HP Turbine horizontal flange leak and bending of some of the LP Turbine coupling bolts at the uprate power. The required bolting changes were accomplished during the R26 refueling.

The OEM evaluation of the main generator and generator support systems showed the generator and supporting systems are capable of supporting the full 7.4% uprate, provided the generator is operated within the existing generator capability curve and the Service Water temperature to the hydrogen coolers does not exceed 86°F. Note that the maximum design service water supply temperature is 80°F. If the generator hydrogen cooling valve is full open, the manual bypass valve may need to be throttled open to augment flow to the Hydrogen Coolers to maintain hydrogen gas temperatures within normal operating limits.

B. Why does the Westinghouse design criteria for MS safety valves having 100% normal steam flow capacity not have to be met?

Response:

The Westinghouse design criteria for the MS safety valves to have 100% rated power steam flow capacity was a conservative criterion for construction of new plants. It was for the original design to provide significant margin to ensure that main steam pressure did not exceed 110% of the steam generator shell-side design pressure. This original design criterion no longer applies to the uprated condition as long as the B31.1 code requirement for protection of the steam generator shell is met. The safety valves provide only approximately 98.6% of the maximum uprated steam flow, however, they continue to meet the over pressure protection requirement (as stated in Sections 4.2.4.1.1 and 6.2 of the Licensing Report).

C. Justify how we can operate the FW heaters beyond their design rating and summarize the inspection results.

Response:

Based on feedwater heater drain data taken from heat balances developed for a 7.4% power uprate, the evaluation identified the potential that the current power (1650 MWt) flow and pressure conditions (rho V squared) associated with the cascading inlet drain lines to some feedwater heaters (11, 12, and 15) exceeded the Heat Exchange Institute (HEI) recommended values. At uprated conditions, only the predicted rhoV2 values for feedwater heater 11 show any appreciable increase. Exceeding the HEI recommended values on these feedwater heaters can potentially result in increased erosion to the inlet nozzles and tubes at the area of the drain inlets. Ultrasonic inspections of the B Loop LP feedwater heaters (11/12B and 13B) and the A and B Loop HP feedwater heaters (14A/B and 15A/B) were performed during the R26 refueling outage in April 2003. These inspections did not identify any evidence of abnormal tube erosion in the areas associated with the heater drain inlets. Since the heater inspections did not identify any areas of concern in the feedwater heater drain inlet areas, the feedwater heaters are considered acceptable to support the uprate to 1772 MWt. Additional inspections will be performed following uprate to identify any subsequent tube erosion and degradation (see LAR 195 Commitment Number 4).

D. Explain why the bolting changes were required for the HP and LP turbines and conclude why the upgraded bolts are acceptable.

Response:

The turbine generator vendor, Siemens Westinghouse, recommended changing out selected HP Turbine horizontal flange bolts and LP turbine coupling bolts to a higher strength material to minimize the potential of a HP turbine horizontal flange leak and bending of some of the LP turbine coupling bolts at the uprate power. Upgraded bolts were deemed acceptable due to following the vendor's recommendations and installation instructions.

E. Explain the regulatory basis of the FAC program.

Response:

NRC Generic Letter 89-08 requested verification that formalized procedures or administrative controls have been implemented to ensure continued implementation of an erosion/corrosion monitoring program. KNPP subsequently committed to initiate and issue procedures and administrative controls for a Pipe Inspection Program. KNPP routinely performs inspections to detect piping degradation caused by Flow Accelerated Corrosion (FAC) or other erosion/corrosion mechanisms. The EPRI software "CHECMATE" is currently used for analyzing pipe inspection data (see LAR 195 Commitment Number 2).

F. Explain the basis of the FW control valve Cv.

Response:

To ensure stable feedwater control with constant speed feedwater pumps, the pressure drop across the FRVs at 100% power should be approximately 1.5 times the feedwater system dynamic losses from the feedwater pump discharge through the steam generators. In addition, to meet the requirements for load rejection, adequate margin should be available in the FRVs at full load conditions to permit a condensate and feedwater system delivery of 96% of rated flow with a 100 psi pressure increase above the full load pressure with the FRVs fully open. A hydraulic flow model of the Condensate and Feedwater systems was used to evaluate system performance under normal and transient conditions. This model was used to evaluate the FRV Cv and determine the expected pressure drop across the FRVs under uprated feedwater flow conditions. The analysis used in the hydraulic model indicated that the current valve Cv would be undersized and would not be adequate to support power uprate normal and abnormal operational requirements. A FRV trim with a larger Cv was chosen which met the flow and pressure drop requirements above (see LAR 195 Commitment Number 3).

2. The Licensee's response to RAI question 4 (November 5, 2003 Letter) stated that the requirements associated with borating the RCS to cold shutdown, utilizing the RWST, with and without letdown in service is being evaluated for any required procedure changes. This evaluation, and update of any required procedures, will be completed prior to implementation of the SUR. This is considered part of our Regulatory Commitment #12. The NRC staff needs to know the results of this evaluation before it can issue a safety evaluation on the acceptability of this amendment request. Please submit a summary of the results of this evaluation.

Response:

A calculation has been performed to demonstrate that following the uprate to 1772 MWt, the Appendix R requirement requiring the ability to borate the RCS to the hot shutdown (HSD) Xenon free minimum technical specification required shutdown margin (SDM) within 24 hours using the RWST as the boration source continues to be satisfied. KNPP assumes letdown can be restored during response to a fire however, the calculation demonstrates that with or without letdown established, the RWST volume required to borate the RCS to the required shutdown margin can be added well within 24 hours. In addition, the calculation demonstrates that if letdown is not established, sufficient Pressurizer volume exists to allow the required RWST volume to be added and still remain within the available Pressurizer level indication. No changes to the existing Appendix R shutdown procedures are required for the stretch uprate.

3. The NRC staff has some additional questions concerning the stretch power uprate.

   A. From WCAP 16040-P, page 8-7 "Most calculated MS flow velocities at power uprate do not exceed the previously accepted recommended maximum values for continuous operation." What are calculated and acceptable values?

Response:

The calculated steam velocities are tabulated in Calculation C11386, Main Steam, and are attached. On Tables 5A – 8B, (Enclosure 3) the Stretch Uprate column is labeled as HB107. The recommended maximum value is 250 feet per second, from a Stone and Webster standard practice.

   B. How did/will the licensee ensure new Feed Reg Valve (FRV) Trim is acceptable?

Response:

Following installation, post maintenance testing showed acceptable performance. This included stroke testing, and pressure drop and valve position monitoring. Current operation has shown that the FRV is performing as designed. Additional monitoring will be performed during the stretch uprate power ascension to ensure performance remains as expected.

   C. How does the licensee ensure sufficient NPSH for the Feedwater pumps? How much margin does the licensee expect in pump suction pressure?

Response:

The low pressure feedwater heater bypass valve (C-13) opens at a feedwater pump suction pressure of 220 psig to ensure sufficient NPSH. This setpoint remains acceptable for stretch uprate conditions and will not change.

The current and predicted pump values are given in the following table.

Table 3.2.3-3b
Feedwater Pump Parameters (Normal Operation)

Pump Parameters	Design Conditions	6% Uprate Conditions	7.4% Uprate Conditions
Flow (gpm)	8,000	8,700	8,800
Head (ft)	2,130	2,090	2,075
NPSHa (ft)	373	332	310
NPSHr (ft)	105	115	115
BHP	4,800	5,100	5,150
Efficiency	84%	85%	86%

In addition, feedwater pump operation will be monitored during the stretch uprate power ascension to ensure operation remains as expected.

D. Is the Feedwater Isolation Valve (FWIV) acceptable for Stretch Uprate containment response?

Response:

The FWIV is credited in the main steam line break (MSLB) accident analyses when the single active failure (SAF) is the failure of a feedwater regulating valve (FRV) to close. The timing assumption for FWIV closure is 90 seconds. For Stretch Uprate, we have shown that the MSLB FRV SAF cases are acceptable at 0%, 30%, 70%, and 100% power, with the most limiting case being from full power.

E. How is the KNPP Flooding Analysis impacted by the power uprate?

Response:

The plant internal flooding impact was evaluated for the HELBs as a result of the power uprate. The flooding due to moderate energy lines was reviewed and was determined not to be impacted, since the moderate energy system operating conditions (pressure and temperature) do not change so as to result in any new HELBs, and the inventory sources for flooding have not increased. Therefore, the internal flooding evaluation was limited to existing HELBs for the impact on the submergence levels identified in the EQ plan. The FW line break outside containment is the bounding break for determining the maximum flood (submergence) levels. The areas flooded and the associated submergence levels are based on specified break locations determined by the HELB criteria contained in the USAR Appendix 10A and Fluor Power Services letter KPS-6601, Kewaunee Nuclear Power Plant Submergence Elevations-High-Energy Line Breaks. The break locations for the FW lines are not impacted since the locations of peak pipe stress are not impacted and the physical line arrangements (terminal ends, branch connections, etc.) have not changed. These inputs are not impacted and are bounded by the existing evaluation. In addition, the maximum flood levels in all compartments, except the basement levels, are controlled by physical plant features such as doors and curb plates. Therefore, the submergence levels in these areas are not impacted by the power uprate.

Results: The flooding (submergence) levels remain the same and are not impacted as a result of the power uprate.

F. Describe the affect of the power uprate on the KNPP missile analysis.

Response:

Systems that are located inside and outside the reactor containment are potential targets of internally generated missiles. High-pressure Reactor Coolant System equipment, which could be the source of missiles is suitably protected either by the concrete wall enclosing the reactor coolant loops or by the concrete operating floor to block any passage of missiles to the containment walls. The steam drum, which forms an integral part of the steam generator, represents a mass of steel which provides protection from missiles originating in the section of the containment within the shield wall and below the operating floor. A concrete structure is provided over the RCCA drive mechanisms to block any missiles generated from a hypothetical fracture of an RCCA housing.

The missile shield structures inside Containment were analyzed using the conservation of momentum method. The penetration depth was calculated by the Ballistic Research Laboratories Formula for steel and concrete.

Missile protection is designed to the following criteria:

a) The Reactor Containment Vessel is protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe (DBA).
b) The Engineered Safety Features Systems and components required to maintain containment integrity are protected against loss of function due to damage by missiles.

During the detailed plant design, the missile protection concepts necessary to meet these criteria were developed. These concepts are:

a) Components of the reactor coolant system are examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects
b) Missile velocities are calculated considering both fluid and mechanical driving forces, which could act during missile generation.
c) The structural design of the missile shielding takes into account both static and impact loads.
d) The Reactor Coolant System is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe, and designed to stop these missiles.

The types of missiles for which protection is provided are:

- Valve Stems
- Valve bonnets
- Instrument thimbles
- Various types and sizes of nuts and bolts
- Complete RCCA drive mechanisms, or parts thereof
- Piece of pipe
- Pressurizer valves, instrumentation thimbles and heaters

Removable slabs, blocks and partitions were evaluated to determine whether they could become possible missiles. These items were reviewed to assure that the functions of safety related structures and equipment would not be affected during an OBE or tornado.

Removable slabs or plates that were found to be potential missiles are anchored to Class I structures by means of bolts and anchor bolts. Concrete inserts and bolts provided for removing the slabs, are designed with lifting capacities equal to two times their normal design stress capabilities.

Turbine Missiles – It has been determined (Ref. 2) that turbine missiles could cause some localized damage to protective concrete barriers or structures. However, as the maximum turbine overspeed is being maintained below 120% (see Engineering Report section for turbine generator), the effects of a turbine missile after uprate will not result in any significant damage.

The reactor thermal power uprate will not adversely impact the evaluations performed for internally generated missiles or their conclusions. The uprate will not increase the number of postulated missiles that require consideration nor increase the equipment required to safely shutdown the plant following the generation of a missile.

Question F References:

1. "Kewaunee Nuclear Power Plant Updated Safety Analysis Report," Rev. 17, Section 5.2, Containment System Structure Design

2. "Kewaunee Nuclear Power Plant Updated Safety Analysis Report," Rev. 17, Appendix B9, Turbine Missile Effects

G. Describe the affect of the power uprate on the Turbine overspeed trip settings (reference WCAP 16040-P, p 8-12).

Response:

Results indicate that with the present trip settings, the unit would overspeed to 121% at the uprated conditions, which slightly exceeds the 120% overspeed design. Therefore, an adjustment to the overspeed trip setting is recommended. The recommended uprate settings as stated in the Siemens-Westinghouse report are as follows:

OVERSPEED TRIP	CURRENT SETPOINT	NEW SETPOINT
MECHANICAL	111%	109%
EHC	111.5%	109.5%
ROST	111.5%	109.5%

The mechanical overspeed trip setting already met the new setpoint requirement and did not require any adjustment. The electro-hydraulic control (EHC) and redundant overspeed trip (ROST) setpoints were adjusted during the April 2003 refueling outage to meet the new setpoint requirements (see LAR 195, Regulatory Commitment Number 6).

H. From WCAP 16040-P, the licensee stated, "The exhaust end loading at the uprated conditions increases. At this loading the exhaust flow guide bolting may be affected. If desired, upgraded bolting can be installed." What actions did the licensee do regarding turbine exhaust flow guide bolting?

Response:

Siemens Westinghouse (TG Vendor) issued AIB 8804 which discusses low pressure turbine exhaust flow guide bolt failures due to stress corrosion and fatigue cracking. KNPP has not implemented the prescribed recommendation of Westinghouse's AIB 8804 for the following reason.

Every other refueling outage, an LP turbine is disassembled, during which an evaluation is performed on the top half bolting of the flow guides (which must be removed for disassembly), and a sample of the bottom half bolting. This evaluation reveals any bolt deterioration due to corrosion and/or erosion that may have occurred during operation, and determines the need for bolt replacement. If an LP is not disassembled during the outage, a crawl-through is performed and the bolts are subject to a visual inspection that may identify the need for bolt replacement. It is much more economical to perform these inspections than to do the bolt replacement modification. KNPP has not had an exhaust flow guide bolt failure, and we are confident that our inspection and maintenance process is satisfactory in minimizing the potential for future bolt failures

4. In the aftermath of the Three Mile Island, Unit 2, accident the NRC requested, as a part of NUREG 0737, an evaluation of the auxiliary feedwater system. Please describe the basis for the present condensate storage tank level found in KNPP technical specifications and describe how KNPP conforms to the Standard Review Plan, section 10.4.9, branch technical position ASB10-1, item B.1

Response:

Condensate Storage Tank Inventory

KNPP's original Technical Specifications required the condensate storage tanks (CST's) to contain 75,000 gallons of water. The specified minimum water supply in the condensate storage tanks (75,000 gallons) was sufficient for two hours at hot shutdown plus cooling of the plant to the cold shutdown condition and maintenance of this condition for an additional four hours. Unlimited replenishment of the condensate storage supply is available from Lake Michigan. Post Three Mile Island, Unit 2, (TMI-2) accident the NRC staff[1] acknowledged this CST level requirement in the staffs' description of the KNPP AFW system.

Subsequent to the NRC staff's description of the AFW system at Kewaunee, several technical specification (TS) changes have occurred to the required CST volume. Starting at 75,000 gallons the TS was changed to 30,000 gallons in 1983, then changed to the present requirement of 39,000 gallons in 1993.

KNPP license amendment 49[2] changed the CST volume requirement from 75,000 gallons to 30,000 gallons. Wisconsin Public Service Corporation (WPSC) requested[3] a level reduction from 75,000 to 10,000 gallons. The bases for 10,000 gallons was the time required for the operators to switch the auxiliary feedwater (AFW) pumps' suction from the CST's to the service water system supply (i.e. Lake Michigan). In the NRC staff's safety evaluation for amendment 49, the staff agreed with WPSCs rational for the 10,000 gallons but stated that post TMI-2 short and long term recommendations GS-5 and GL-3, identified in NUREG 0611, require that the AFW system be capable of operation for at least 2 hours following a complete loss of all alternating-current (a-c) power sources [what is now called a station blackout]. The TS basis, included with this amendment, stated that this specified minimum water supply in the condensate storage tanks was sufficient for ninety minutes at hot shutdown plus a suitable margin to prevent loss of net positive suction head prior to switching suction to the service water system.

KNPP license amendment 97[4] changed the minimum CST volume from 30,000 gallons to 39,000 gallons, the current CST volume requirements. In this SE the NRC staff stated that in a November 20, 1990, Safety Evaluation (SE) addressing Kewaunee's implementation of the Station Blackout (SBO) Rule, the NRC staff recommended a TS amendment to increase the minimum required CST inventory. The licensee determined that a minimum of 39,000 gallons in the CSTs satisfies the 4-hour SBO coping duration requirement to remove heat. Additionally, the KNPP TS basis states that this specified minimum water supply in the condensate storage tanks is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site-specific station blackout coping duration requirements.

Condensate Storage Tank Response References:

[1] Letter from Darrell G. Eisenhut (NRC) to Eugene R. Mathews (WPSC), "NRC Requirements for Auxiliary Feedwater Systems at Kewaunee Plant," dated September 21, 1979, enclosure, 1 page 2.
[2] Letter from Marshall Grotenhuis (NRC) to C.W. Geisler (WPSC), dated April 28, 1983.
[3] Letter from C.W. Geisler (WPSC) to H.R. Denton (NRC), "Proposed Amendment No. 51 to the KNPP Technical Specifications," dated December 20, 1982
[4] Letter from Allen G. Hansen (NRC) to C.A. Schrock (WPSC), "Amendment NO. 97 to Facility Operating License NO. DPR-43 (TAC NO. M84062)," dated February 1, 1993.

Auxiliary Feedwater System requirements

NUREG 0800, Standard Review Plan, "Section 10.4.9, "Auxiliary Feedwater Systems," (AFW) Rev. 2, dated July 1981, contain the NRC acceptance criteria in question for the KNPP AFW system. Section II, item 5.b states:

II. ACCEPTANCE CRITERIA
Acceptability of the design of the auxiliary feedwater system, as described in the applicant's safety analysis report (SAR), is based on specific general design criteria and regulatory guides. Listed below are the specific criteria used in this SRP section as they relate to the AFWS.
5. General Design Criteria 34 and 44, to assure:
b. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.) Branch Technical Position ASB 10-1 as it relates to AFW pump drive and power supply diversity shall be used in meeting these criteria.

Standard Review Plan Branch Technical Position ASB 10-1, item B.1 states:

The auxiliary feedwater system should consist of at least two full-capacity independent systems that include diverse power sources.

10 CFR 50 Appendix A, "General Design Criteria (GDC) states the following for GDC 34 and 44.

Criterion 34--Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Criterion 44--Cooling water. *A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.*

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

KNPP accident analysis demonstrates that GDC 34 and 44 are met. NMC has performed accident and transient analysis for all events that credit the AFW system. In these analyses the only accident or transient for which unacceptable results are achieved with only one AFW pump is the loss of normal feedwater accident (LONF). The LONF event requires two AFW pumps to achieve acceptable results; this still meets the GDC 34 and 44 requirements.

KNPP's initial license required two AFW pumps to be operable. However, due to scenarios where a depressurization of the steam generator (e.g. feed line break or steam line break (SLB)) were to occur concurrent with a single active failure, three AFW pumps were needed to provide feedwater to the intact steam generator.[1] In the safety evaluation for the addition of the three AFW pump operability requirement the NRC staff[2] concurred with this justification. This license amendment also added the requirement that if one AFW pump was out-of-service the licensee had 72 hours to return the pump to an operable status or the reactor shall be shutdown and cooled below 350°F using normal operating procedures.

After the TMI-2 accident the NRC issued requirements[3] for the auxiliary feedwater system for the Kewaunee Nuclear Power Plant identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident. This letter contains two type of requirements; 1) generic requirements applicable to most Westinghouse-designed operating plants, and 2) plant-specific requirements applicable only to Kewaunee. Included in the letter is recommendation GS-1 (generic, short-term) that stated that the licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The letter recommended 72 hours with a subsequent action time of 12 hours. This letter did not state a basis for this recommendation.

The WPSC response[4] to this recommendation (GS-1) stated that no proposed modification to the KNPP TS associated with the AFW pumps is warranted. WPSC stated that the AFW system at KNPP is designed in accordance with general design criteria (GDC) 41 among others, the redundant motor driven pumps are supplied power from safeguards buses, two of the three AFW pumps are required to be available, and if two AFW pumps are unavailable for 48 hours the plant is required to be shutdown in a shorter time period then the NRC recommended. Therefore, no action is warranted. The NRC responded[5] that this was unacceptable and that WPSC should propose a TS modification. WPSC responded[6] stating the NRC's recommendation to require three auxiliary feedwater pumps to be operable whenever the reactor coolant system is above 350°F is not provided with technical justification and that KNPP's current TS comply with 10CFR50.36 for what needs to be included in TS. WPSC further stated that when technical justification is received it would be reviewed and responded to accordingly.

Subsequently, NUREG 0737, "Clarification of TMI Action Plan Requirements," was issued (October 31, 1980[7]) which contains an item related to an evaluation of the AFW system; item II.E.1.1, "AFW System Evaluation." This item included an action to perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance. For the KNPP this item was closed out by an NRC safety evaluation[8] (SE) concluding that the licensee's response to short term recommendation, GS-1, was acceptable.

In the NRC's SE for GS-1, two reasons were stated for the justification for requiring all three AFW pumps to be included in the TS. Although not specifically stated it appears these reasons are also the justifications for two full-capacity independent systems that include diverse power sources. These reasons are[9]:

a. An order of magnitude increase in system reliability is noticed for a three pump system versus a two pump system. And with the Kewaunee specification the system must be considered as having only two pumps since a third pump can be down for maintenance for an indefinite period, and

b. If the turbine driven pump is the one not available, then the system will not have power diversity and the plant will not be able to remove decay heat in the event of a complete loss of all AC power sources [station blackout].

The KNPP TS were modified to include three AFW pumps but not for the same reasons the NRC stated. Wisconsin Public Service Corporation (WPSC), because of concerns identified during an NRC vital area analysis, agreed[10] to modify KNPP TS to include all three AFW pumps, which was completed when KNPP TS amendment 45 was implemented. The basis for the three AFW pumps was stated in the NRC's SE[11] as being necessary to meet the circumstances following a rupture of a steam pipe situation. Subsequently, in the NRC's letter dated December 16, 1982, the NRC closed out generic short-term recommendation GS-1.

To ensure adequate flow capability for a station blackout the turbine driven AFW pump was required to be full capacity, which is true for all accident analyses for the present KNPP operating power level. At uprated power (1772 MWth) the only accident where the turbine driven AFW pump is not full capacity is the loss of normal feedwater (LONF) event. Additionally, if the reactor coolant pumps, which are assumed to supply 8 MWth to the reactor coolant system, were tripped, the turbine driven AFW pump would supply adequate flow to achieve acceptable results in the LONF analysis. For the station blackout (SBO) analysis, the rational given for the power diversity (i.e. the turbine driven AFW pump has full capacity) is still met for uprated power. In a SBO event the reactor coolant pumps are tripped.

Therefore, at 1772 MWth, the KNPP TS still requires three AFW pumps to be operable, meets the requirements of GDC's 34 and 44. Additionally, for events where the reactor coolant pumps are tripped, such as station blackout, KNPP two full capacity systems with diverse power sources, meeting the intent of the standard review plan branch technical position guidelines for diverse power sources.

Auxiliary Feed Water System References:

[1] Letter from E.R. Mathews (WPSC) to S.A. Varga (NRC), "Auxiliary Feedwater System," dated October 1, 1981.
[2] Letter from Marshall Grotenhuis (NRC) to Eugene R. Mathews (WPSC), dated June 2, 1982, KNPP License Amendment 45.
[3] Letter from Darrell G. Eisenhut (NRC) to Eugene R. Mathews (WPSC), "NRC Requirements for Auxiliary Feedwater Systems at Kewaunee Plant," dated September 21, 1979.
[4] Letter from E.R. Mathews (WPSC) to Darell G. Eisenhut (NRC), "NRC Requirements for Auxiliary Feedwater System at Kewaunee Plant," dated October 30, 1979.
[5] Letter from Steven A. Varga (NRC) to Eugene R. Mathews (WPSC), Kewaunee Nuclear Power Plant: Additional Information Requirements for the Auxiliary Feedwater System," dated August 18, 1980.
[6] Letter from E.R. Mathews (WPSC) to Steven A. Varga (NRC), "Additional Information for Auxiliary Feedwater System," dated October 17, 1980.
[7] Introductory letter from Darrell G. Eisenhut (NRC) to all licensees, "Post TMI-Requirements," dated October 31, 1980.
[8] Letter from Steven A. Varga (NRC) to C.W. Giesler (WPSC), "Kewaunee Nuclear Power Plant – Safety Evaluation Input on the Implementation of Recommendations for the Auxiliary Feedwater Systems," dated December 16, 1982.
[9] Letter from Steven A. Varga (NRC) to C.W. Giesler (WPSC), "Kewaunee Nuclear Power Plant – Safety Evaluation Input on the Implementation of Recommendations for the Auxiliary Feedwater Systems," dated December 16, 1982, safety evaluation page 2.
[10] Letter from E.R. Mathews (WPSC) to S.A. Varga (NRC), Auxiliary feedwater System," dated October 1, 1981
[11] Letter from Marshall Grotenhuis (NRC) to Eugene R. Mathews (WPSC), dated June 2, 1982

5. Describe the results of the impact of the power uprate on the GL 96-06 results.

Response:

A) <u>GL 96-06 Thermal Over pressurization of Isolated Water-Filled Piping Systems</u>:

Generic Letter 96-06 was issued to address (among other concerns) the concern of thermally induced over pressurization of isolated water-filled piping sections in containment that could jeopardize the ability of accident-mitigating systems to perform their safety functions and also could lead to a breach of containment integrity via bypass leakage.

The potential for a thermally induced over-pressurization of isolated water-filled piping systems in containment is not impacted by the uprate since:

1) The volumes of isolated water-filled piping sections in containment are not being changed by uprate conditions
2) The design basis accident (DBA) containment temperature and pressure response values for the 7.4% uprate conditions are bounded by the values that were used for the current KNPP GL 96-06 thermal over pressurization analysis.

Therefore the current GL 96-06 analysis and operability determination for thermally induced over pressurization of isolated water-filled piping sections inside containment remain bounding and valid for operation at stretch power uprate conditions.

B) <u>GL 96-06 Two-Phase Flow in Service Water System (SWS) Discharge Piping for Containment Fan Coil Units (CFCUs)</u>:

The CFCU's operation for accident conditions was based on the design containment temperature and pressure limits. The limiting condition for the CFCUs is the SW exit temperature from the cooling coils. In order to avoid two-phase flow in the lines downstream of the CFCUs, the CFCU SWS exit temperature must be limited such that the corresponding saturation pressure is below the actual system pressure in these lines. The GL 96-06 concern is that two-phase flow could result in higher back- pressure conditions that could reduce the SW flow to the CFCU's below the minimum required flow of 800 gpm during accident conditions.

Based on the current operability analysis and SWS configuration the saturation pressure for the maximum SW temperature in the exit line from the CFCUs is below the static pressure at this location for all containment temperatures up to and including the 268°F design temperature. The static pressure, based on the current return line arrangement, is sufficient to prevent flashing and two-phase flow in these lines.

The potential for flashing and two-phase flow of the SW in the outlet/drain lines from the CFCUs was evaluated for stretch power uprate conditions. The current GL 96-06 analysis was based on containment being at the containment design pressure and temperature limit values, no CFCU tube side fouling (ideal heat exchanger to maximize heat input to SWS), and a CFCU SWS flow of 800 gpm.

The containment analysis for the 7.4% uprate shows that the maximum containment temperature and pressure are less than the containment pressure and temperature design limits. The power uprate containment conditions are therefore bounded by the conditions assumed in the GL 96-06 two- phase flow analysis.

The current GL 96-06 analysis for two phase flow conditions in SWS piping downstream of the CFCU's is therefore bounding and remains valid for stretch power uprate operation.

C) <u>CFCU SW Water hammer</u>:

Generic Letter 96-06 was issued for licensees to determine if the SWS lines to and from the CFCU's were susceptible to water hammer during accident conditions.

An evaluation of the current GL 96-06 operability was performed for the power uprate. The potential for water hammer is not impacted by the power uprate since:

1) The containment temperature and pressure values for the DBA's for the 7.4% uprate conditions are bounded by the values that were used for the current KNPP GL 96-06 operability analysis.
2) Based on sensitivity studies performed in the current operability analyses different containment temperature profiles were shown to not significantly impact the analysis results.

GL 96-06 analyses have been performed for KNPP by a vendor under the vendor's quality assurance program using the EPRI methodology. The new GL 96-06 analyses are currently being reviewed by NMC and may need to be revised. The new analysis results bound the stretch power uprate plant operating conditions and demonstrate that the SWS is operable for the GL 96 –06 water hammer issue. The new analyses using the EPRI methodology also show that power uprate will have no impact on the GL 96-06 water hammer results.

Due to the need to complete our review of the vendor analysis and the potential need for revision to the vendor analysis, NMC will provide NRC with a status update and schedule for submittal of the resolution of GL 96-06 water hammer issues by April 2, 2004.

Therefore based on the evaluation performed for stretch power uprate conditions of the current GL 96-06 operability analysis and the results from the new analysis with the EPRI methodology, the SWS GL 96-06 water hammer operability analysis is bounding and valid for operation at stretch power uprate conditions.

NRC SPSB Branch Questions

1. Please describe the assumptions made regarding the behavior of drops formed by the break flow in high energy line break calculations done for structural analysis (Section 8.5 of WCAP 16040-P) and equipment qualification calculations (Section 8.9 of WCAP 16040-P).

Response:

For power uprate structural analysis (section 8.5 of WCAP 16040-P) it was concluded that the initial conditions for the current structural analysis that is based on a high energy line break from no load conditions (1020 psia main steam system pressure) are the same initial condition assumed for the stretch power uprate.

For the power uprate equipment qualification calculations (section 8.9 of WCAP 16040-P) the high energy line break (HELB) mass and energy releases were conservatively determined with no water entrainment (no water drops) for the entire break spectrum included in the analysis. This assumption maximizes the temperature of the atmosphere in the room and thus the temperature of the equipment for environmental qualification.

2. What single failure assumptions are made in the analyses to support TS 3.3.c.1.A.3?

Response:

The single failure of one emergency diesel generator to start and load one of the two trains of containment engineered safeguards equipment (one train of containment engineered safeguards equipment consists of one train of containment spray and two containment fan coil units) is the single failure assumption in the containment integrity analyses This single failure assumption supports TS 3.3.c.1.A.3.

3. Please verify that the assumption that the break mitigation hardware reduces the break discharge to 50% of the no load flow in the cubicle pressurization analyses is consistent with the current Kewaunee licensing basis. If it is not, please provide a justification for this assumption.

Response:

The current Kewaunee licensing basis cubicle pressurization analysis, documented in the Kewaunee Nuclear Power Plant Updated Safety Analysis Report, Rev. 18, Appendix 10A, Postulated Pipe Failure Analysis, assumes that the break mitigation hardware reduces the pipe break discharge flow rate to 50 percent of the no load flow rate.

NRC SRXB Branch Questions

1. In the November 5, 2003 response, number 34, NMC stated that calculations demonstrate that even if upper plenum injection is established as late as 18 hours after the LOCA (SBLOCA), the vessel boron concentration will still be 4 weight % under the boron precipitation point, assuming a saturation pressure of 35 psia. Other nuclear plants have stated that they will reach the boron saturation point in approximately 6-9 hours (Adams Accession NO. ML032720538). Please discuss this inconsistency and any actions necessary to ensure boric acid precipitation is not a concern.

Response:

As explained in our responses to RAI #'s 33 and 34, the emergency operating procedures (EOP) direct the operator to verify low-head recirculation (via the vessel upper head injection lines) at RCS pressures below 150 psig. The subject calculations conservatively assume a boric acid solubility limit that corresponds to an RCS at saturated conditions at 20 psig (35 psia). This boric acid solubility limit used at 20 psig was 38 w/o and was taken from experimental data (with 4% uncertainty margin). The boric acid solubility limit typically used for non-UPI plants that must consider the LBLOCA scenario (full RCS depressurization) is 23.53 w/o with 4% uncertainty margin. The difference between 38 w/o and 23.53 w/o solubility limit explains the cited difference between the Kewaunee calculation and the calculations for non- UPI plants that result in 6-9 hours (Adams accession No. ML03270538).

KNPP is a two-loop Westinghouse reactor plant with upper plenum injection (i.e., the low head safety injection (SI) pumps inject emergency core cooling water directly into the reactor vessel upper plenum). The shutoff head of the low head SI pumps is approximately 150 psig. Therefore, as soon as the reactor coolant system (RCS) pressure falls below about 150 psig, then the low head SI pumps would be capable of injecting cooling water into the RCS upper plenum. For a large-break loss of coolant accident (LOCA) the reactor coolant system (RCS) depressurizes quickly to below the shutoff head of the low head SI pumps and therefore, ECCS flow is quickly injected into the reactor vessel upper plenum directly above the reactor core at high flow rates to address the potential boron precipitation concern.

For small-break LOCAs the RCS does not immediately depressurize. However, the KNPP Integrated Plant Emergency Operating Procedures (IPEOPs) direct the operators to perform a post-LOCA cooldown and depressurization of the RCS. It is expected that the initial operator response to a small-break LOCA would be completed and an RCS cooldown started within about one hour. The cooldown rate is limited to a maximum of 100 degrees F per hour due to reactor vessel integrity concerns. It is expected that the RCS would be cooled down and depressurized to well below the low head SI pumps' shutoff head of 150 psig within a total time of approximately 6 to 8 hours. This will result in upper plenum injection of ECCS water directly into the reactor vessel above the reactor core, which addresses the potential boron precipitation concern for small-break LOCAs.

2. The NRC requests additional information concerning the elimination of the DNBR rod bow penalty for application to KNPP Cycle 26. Specifically, the NRC requests detailed information is provided about the way the FDH burndown was calculated during the cycle and the way the corresponding DNBR credit was calculated.

Response:

Based on discussions with NRC, NMC has decided to withdraw its request for approval of the non-standard rod bow methodology that was used to evaluate the rod bow DNBR penalties at 1772 MWth reactor thermal power. Instead the standard Westinghouse rod bow methodology including the 2.6% rod bow DNBR penalty has been applied to KNPP C 26 core thermal hydraulic analyses at the stretch uprate (SUR) conditions. DNBR margins were shown to be sufficient for C 26 SUR operation with the standard rod bow methodology and standard rod bow penalty of 2.6% applied. Therefore the thermal hydraulic design of the Cycle 26 reload at conditions up to and including the SUR meets the applicable TS and COLR limits and remains bounded by the safety analyses.

NRC SPSB-C Branch Question

1. With regard to the waste gas decay tank rupture and volume control tank rupture analyses, you state that in absence of guidance in RG 1.183, you have assumed that the offsite dose acceptance criterion is the same as given for the fuel handling accident (i.e., 6.3 rem TEDE). However, in the staff's safety evaluation (issued March 17, 2003) for implementation of the alternative source term, this acceptance criterion was not used by the NRC staff in its finding. Kewaunee was originally licensed with a site boundary dose acceptance criterion of 500 mrem whole body for these accidents. Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas Tank System Leak or Failure," which is located in SRP Section 11.3, "Gaseous Waste Management Systems," also presents a dose acceptance criterion of 500 mrem whole body. The curie content of these tanks is controlled so that 10 CFR 20.1301 dose limits to individual members of the public are not exceeded in the event of a release.

   A. Please explain how your assumed acceptance criterion of 6.3 rem TEDE was chosen.
   B. Please explain how your assumed dose acceptance criterion of 6.5 rem TEDE for these accidents is consistent with Kewaunee's original licensing and design basis, and Branch Technical Position ETSB 11-5, or why the difference should be found acceptable.
   C. Please explain how the assumed accident dose acceptance criterion interacts with control of radioactivity in the tanks and the 10 CFR 20 dose limits for individual members of the public.

Response:

A & B. The appropriate KNPP offsite dose limit for the gas decay tank rupture and volume control tank rupture analyses is 0.5 rem TEDE, consistent with the original licensing basis and Branch Technical Position ETSB 11-5. The site boundary dose reported for the gas decay tank rupture and volume control tank rupture is 0.1 rem TEDE, which is below the limit. KNPP is not requesting that a higher limit be accepted.

C. At the time of KNPP's original license the 10CFR20 limit for individual members of the public was 0.5 rem in a year. This earlier version of 10CFR20 is also the basis for the Branch Technical Position ETSB 11-5 limit of 0.5 rem for the short-term dose due to a tank failure. Therefore, although the current version of 10CFR20 identifies a limit of 0.1 rem, the KNPP limit of 0.5 rem remains applicable. The calculation of the tank inventories is described in Section 7.6 of the licensing report (Attachment 4 of LAR 195[1]).

NRC EEIB Branch Question:

1. NMC's December 15, 2003 response to NRC item 9 described the setpoint methodology used for the KNPP. The NRC staff requires additional information associated with how channel operability is determined.

Response:

WCAP-15821-P, Rev.0 (Ref. 1) provides the basis for the COLR OTDT and OPDT reactor trip setpoints. Note the following differences in terminology:

1) The Kewaunee technical specifications do not have an allowable value and technical specification setpoints are referred to as limiting safety system settings or instrument setting limits.

2) The Kewaunee Core Operating Limits Report (COLR) does not have an Allowable Value. The Kewaunee COLR setpoints are referred to as setpoints.

The total instrument channel uncertainty [or Channel Statistical Allowance (CSA)] is calculated with all measured and unmeasured instrument channel uncertainties. The total instrument channel uncertainty + margin are subtracted from a defined safety analysis limit (SAL) to determine the Technical Specification or COLR setpoint. A more conservative Kewaunee plant setting is used to prevent a violation of the Technical Specification or COLR setpoint.

KNPP instrumentation for the OPDT indicates in percent of full power reactor coolant loop delta T (ΔT). The span of the instrumentation is 0 – 150%. For the COLR OPDT reactor trip setpoint, WCAP-15821-P, Rev.0 shows a CSA of 4.2% of span (see Table 3-8 (Ref. 1)), a margin of 0.1% of span (see Table 3-8 (Ref. 1)), and the K_4 SAL of 1.16 (see Table 3-10 (Ref. 1)). The corresponding COLR OPDT K_4 is 1.095 (see Table 3-10 (Ref. 1)). This results in a positive margin of 0.065 (1.16 – 1.095) between the SAL and the COLR OPDT Trip Setpoint, which is > the total CSA of 4.2% of span.

Percent of Span to % Full Power ΔT Indicated Value		
% of Span	Span	% Full Power ΔT Indication Value
4.2	150%	6.3
0.1	150%	0.15
4.33	150%	6.5

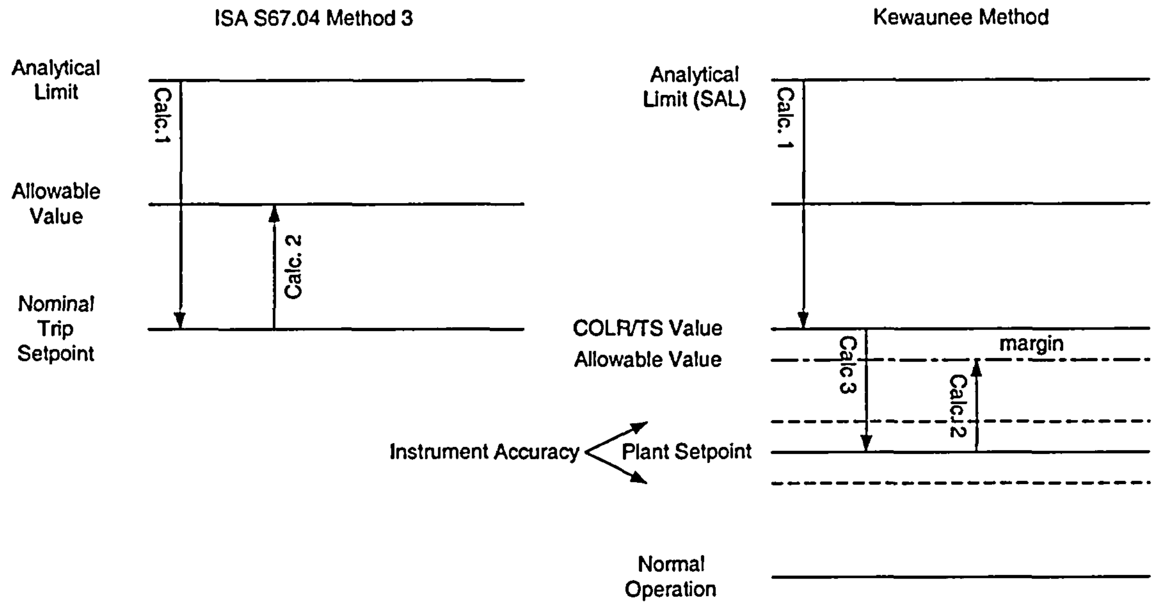
Therefore, as there is margin between the assumed safety analysis limit and the Technical Specification or COLR value, after accounting for all uncertainties for a given protection function, there will be an increase in margin when addressing a subset of the uncertainties.

Using the above information for the OPDT instrument the determination of the setpoint and operability value can be further explained. The SAL is 1.16. This means that the operators would see a value of 116% on their OPDT setpoint indication if this were the OPDT setpoint. To determine the OPDT setpoint the CSA and margin are subtracted from the SAL (1.16 - 0.063 - 0.0015 = 1.0955 [or ≅ 1.095]). This is the value that is entered into the KNPP COLR for the OPDT setpoint (K_4). To determine the plant trip setpoint additional margin is subtracted from the COLR value. This value (margin) is a historical number that has been demonstrated by experience to ensure the COLR value will not be exceeded (0.0125). Adding this margin yields a plant trip setpoint of 1.0825 or 108.25% full power ΔT, as indicated in the control room.

NUREG 1431, Standard Technical Specifications Westinghouse Plants," revision 2 states that there is some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This value needs to be specified in the technical specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value. NMC determines an allowable value but not by the same method used in ISA 67.04 and does not currently utilize this "Allowable Value" for the determination of operability for the KNPP. NMC uses the plant setpoint ± instrument accuracy value for the determination of operability.

For operability and calibration tolerance of the OPDT trip function at KNPP NMC uses the COLR/TS value and the instrument accuracy, respectively (see Figure 1). For example, an instrument and control (I&C) technician is performing a calibration of an OPDT channel. The I&C technician finds the K_4 setpoint to be 1.085. The I&C technician will leave the setpojnt as found because it is within 0.5% span (0.5% span is instrument accuracy or 0.5% x 150 = 0.0075) of plant setpoint of 1.0825. Thus if the I&C technician finds the setpoint to be within 1.0825 ± 0.0075 the setpoint is not adjusted, it is left as is. If the as found setpoint is outside the ±0.5% instrument accuracy the setpoint is adjusted to be within that instrument accuracy band and the channel determined to be inoperable until adjusted within the accuracy band. If an as found setpoint is outside of the instrument accuracy band NMC also performs an evaluation of the condition. If the setpoint is found to be within the KNPP allowable band the instrument is determined to meet its design function.

Figure 1



Calculation 1 (Calc.1) adds/subtracts the measured uncertainties and unmeasured uncertainties plus margin from the analytical limit.
Calculation 2 (Calc.2) adds/subtracts the measured uncertainties.
Calculation 3 (Calc.3) adds/subtracts the measured uncertainties plus a margin (margin may be as low as zero)

Although NMC uses the instrument accuracy in determining operability of the instrument, the plan is to expand the operability band to the KNPP allowable value. Thus the instrument would be considered operable until the setpoint exceeded the KNPP allowable value. Additionally there are some instruments at KNPP that the value for Calc. 3 and/or Calc. 2 has not been determined. For these instruments NMC uses the TS/COLR value, adds/subtracts a historical value that has been shown to prevent exceeding the TS/COLR value and bases operability on the instrument accuracy band.

Thus further margin is added between the analytical limit and the actual plant setpoint.

TS Setting Limits cannot be changed without NRC approval per current processes and regulations.

[1] Letter from Thomas Coutu (NMC) to Document Control Desk (NRC), License Amendment Request 195, Application for Stretch Power Uprate for Kewaunee Nuclear Power Plant," dated May 22, 2003 (Adams Accession NO. ML031530424)

ENCLOSURE 3

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

January 30, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

NMC Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195

KNPP Main Steam Piping Velocities

8 Pages Follow

TABLE 5A

Main Steam Piping Velocities
Steam Generator A to High Pressure Turbine Stop Valve

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			744.25		770.75		785.75		800.75		770.75		785.75		800.75	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator A Outlet to 31.5" OD Piping Inlet	30" OD / 0.9742" min wall	28.0516	8,313	138.6	8,569	142.8	8,398	140.0	8,233	137.2	8,696	144.9	8,522	142.0	8,355	139.2
31.5" OD Piping Inlet to 31.5"OD Piplgn Outlet	31" OD / 1.461" min wall	28.078	8,382	139.7	8,646	144.1	8,470	141.2	8,301	138.3	8,777	146.3	8,598	143.3	8,426	140.4
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,416	140.3	8,683	144.7	8,505	141.8	8,334	138.9	8,815	146.9	8,635	143.9	8,461	141.0
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,417	140.3	8,684	144.7	8,506	141.8	8,335	138.9	8,816	146.9	8,636	143.9	8,462	141.0
30" x 24" Cross to Steam Dump Branch (18" Line to Atmosphere)	30" OD / 0.9742" min wall	28.0516	8,427	140.4	8,695	144.9	8,516	141.9	8,345	139.1	8,827	147.1	8,646	144.1	8,472	141.2
Steam Dump Branch (18" Line to Atmosphere) *to Steam Dump Branch* (18" Line to Condenser)	30" OD / 0.9742" min wall	28.0516	8,508	141.8	8,785	146.4	8,602	143.4	8,425	140.4	8,922	148.7	8,735	145.6	8,556	142.6
Steam Dump Branch (18" Line to Condenser) to 20" Equalizing Line	30" OD / 0.9742" min wall	28.0516	8,340	139.0	8,621	143.7	8,430	140.5	8,247	137.4	8,762	146.0	8,567	142.8	8,381	139.7
20" Equalizing Line to 8" Branch to MSRs	30" OD / 0.9742" min wall	28.0516	8,375	139.6	8,660	144.3	8,467	141.1	8,281	138.0	8,803	146.7	8,606	143.4	8,417	140.3
8" Branch to MSRs to HP Turbine Stop Valve	30" OD / 0.9742" min wall	28.0516	7,832	130.5	8,103	135.0	7,908	131.8	7,721	128.7	8,243	137.4	8,044	134.1	7,854	130.9

TABLE 5B

Main Steam Piping Velocities
Steam Generator B to High Pressure Turbine Stop Valve

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			736.75		763.25		778.25		793.25		763.25		778.25		793.25	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator B Outlet to 31.5" OD Piping	30" OD / 0.9742" min wall	28.0516	8,834	147.2	9,103	151.7	8,919	148.7	8,743	145.7	9,238	154.0	9,052	150.9	8,872	147.9
31.5" OD Piping Inlet to 31.5"OD Piplgn Outlet	31" OD / 1.461" min wall	28.078	8,919	148.7	9,199	153.3	9,009	150.2	8,827	147.1	9,338	155.6	9,146	152.4	8,961	149.3
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,959	149.3	9,242	154.0	9,050	150.8	8,866	147.8	9,383	156.4	9,188	153.1	9,001	150.0
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,960	149.3	9,243	154.0	9,051	150.9	8,867	147.8	9,384	156.4	9,189	153.2	9,003	150.0
30" x 24" Cross to Steam Dump / MSR Branch (18" Stm Dump to Condenser)	30" OD / 0.9742" min wall	28.0516	8,972	149.5	9,256	154.3	9,064	151.1	8,879	148.0	9,397	156.6	9,202	153.4	9,015	150.2
Steam Dump / MSR Branch (18" Line to Condenser) to Steam Dump Branch (18" Line to Atmosphere)	30" OD / 0.9742" min wall	28.0516	8,775	146.3	9,060	151.0	8,862	147.7	8,672	144.5	9,205	153.4	9,003	150.1	8,810	146.8
Steam Dump Branch (18" Line to Atmoshpere) to 20" Equalizing Line	30" OD / 0.9742" min wall	28.0516	8,801	146.7	9,089	151.5	8,889	148.1	8,697	145.0	9,235	153.9	9,031	150.5	8,836	147.3
20" Equalizing Line to HP Turbine Stop Valve	30" OD / 0.9742" min wall	28.0516	8,839	147.3	9,131	152.2	8,928	148.8	8,734	145.6	9,279	154.6	9,073	151.2	8,875	147.9

TABLE 6A

Atmospheric Dump Valve Piping Velocities
Steam Generator A

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			740.50		767.00		782.00		797.00		767.00		782.00		797.00	
Piping Description	**Pipe Size/Schedule**	**d Inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator A Outlet to 31.5" OD Piping Inlet	30" OD / 0.9742" min wall	28.0516	8,572	142.9	8,835	147.2	8,657	144.3	8,487	141.4	8,966	149.4	8,786	146.4	8,612	143.5
31.5" OD Piping Inlet to 31.5"OD Pipign Outlet	31" OD / 1.461" min wall	28.078	8,648	144.1	8,920	148.7	8,738	145.6	8,562	142.7	9,056	150.9	8,870	147.8	8,691	144.9
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,685	144.7	8,959	149.3	8,775	146.3	8,598	143.3	9,096	151.6	8,909	148.5	8,729	145.5
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,686	144.8	8,960	149.3	8,776	146.3	8,599	143.3	9,097	151.6	8,910	148.5	8,729	145.5
30" x 24" Cross to Steam Dump Branch (18" Line to Atmosphere)	30" OD / 0.9742" min wall	28.0516	8,696	144.9	8,972	149.5	8,787	146.5	8,609	143.5	9,109	151.8	8,921	148.7	8,740	145.7
From SG A Main Header to FCV468A Takeoff	18" OD / Sch 80	16.124	11,967	199.4	12,356	205.9	12,096	201.6	11,846	197.4	12,548	209.1	12,284	204.7	12,031	200.5
FCV468A Takeoff to FCV468B Takeoff	18" OD / Sch 80	16.124	8,123	135.4	8,398	140.0	8,215	136.9	8,040	134.0	8,534	142.2	8,349	139.1	8,170	136.2
FCV468B Takeoff to FCV468C Takeoff	18" OD / Sch 80	16.124	4,067	67.8	4,205	70.1	4,113	68.6	4,025	67.1	4,273	71.2	4,180	69.7	4,090	68.2
18" Header to FCV468A	8" OD / Sch 80	7.625	18,162	302.7	18,777	312.9	18,368	306.1	17,977	299.6	19,081	318.0	18,666	311.1	18,267	304.5
18" Header to FCV468B	8" OD / Sch 80	7.625	18,185	303.1	18,802	313.4	18,392	306.5	18,000	300.0	19,108	318.5	18,691	311.5	18,291	304.9
18" Header to FCV468C	8" OD / Sch 80	7.625	18,191	303.2	18,809	313.5	18,399	306.6	18,005	300.1	19,115	318.6	18,698	311.6	18,297	305.0

TABLE 6B

Atmospheric Dump Valve Piping Velocities
Steam Generator B

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			740.50		767.00		782.00		797.00		767.00		782.00		797.00	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator B Outlet to 31.5" OD Piping	30" OD / 0.9742" min wall	28.0516	8,572	142.9	8,835	147.2	8,657	144.3	8,487	141.4	8,966	149.4	8,786	146.4	8,612	143.5
31.5" OD Piping Inlet to 31.5"OD Piplgn Outlet	31" OD / 1.481" min wall	28.078	8,649	144.2	8,922	148.7	8,739	145.6	8,563	142.7	9,057	150.9	8,871	147.9	8,693	144.9
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,687	144.8	8,961	149.4	8,777	146.3	8,600	143.3	9,098	151.6	8,911	148.5	8,730	145.5
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,688	144.8	8,963	149.4	8,778	146.3	8,601	143.3	9,099	151.7	8,912	148.5	8,731	145.5
30" x 24" Cross to Steam Dump / MSR Branch (18" Stm Dump to Condenser)	30" OD / 0.9742" min wall	28.0516	8,698	145.0	8,974	149.6	8,789	146.5	8,611	143.5	9,111	151.9	8,923	148.7	8,743	145.7
Steam Dump / MSR Branch (18" Line to Condenser) to Steam Dump Branch (18" Line to Atmosphere)	30" OD / 0.9742" min wall	28.0516	8,790	146.5	9,075	151.3	8,884	148.1	8,701	145.0	9,217	153.6	9,023	150.4	8,837	147.3
From SG B Main Header to 18" Atmospheric Dump Header	18" OD / Sch 80	16.124	12,007	200.1	12,399	206.7	12,137	202.3	11,885	198.1	12,594	209.9	12,328	205.5	12,072	201.2
18" Atmospheric Dump Header to FCV478C Takeoff	18" OD / Sch 80	16.124	4,047	67.5	4,183	69.7	4,093	68.2	4,006	66.8	4,251	70.8	4,159	69.3	4,070	67.8
18" Atmospheric Dump Header to FCV478B Takeoff	18" OD / Sch 80	16.124	8,095	134.9	8,366	139.4	8,186	136.4	8,012	133.5	8,501	141.7	8,317	138.6	8,141	135.7
FCV478B Takeoff to FCV478A Takeoff	18" OD / Sch 80	16.124	4,052	67.5	4,189	69.8	4,098	68.3	4,011	66.8	4,256	70.9	4,164	69.4	4,075	67.9
18" Header to FCV478C	8" OD / Sch 80	7.625	18,104	301.7	18,712	311.9	18,307	305.1	17,919	298.7	19,013	316.9	18,602	310.0	18,207	303.4
18" Header to FCV478B	8" OD / Sch 80	7.625	18,120	302.0	18,730	312.2	18,324	305.4	17,936	298.9	19,032	317.2	18,620	310.3	18,224	303.7
18" Header to FCV478A	8" OD / Sch 80	7.625	18,126	302.1	18,737	312.3	18,331	305.5	17,941	299.0	19,040	317.3	18,626	310.4	18,230	303.8

TABLE 7A

Condenser Dump Valve Piping Velocities
Steam Generator A

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			740.50		767.00		782.00		797.00		767.00		782.00		797.00	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator A Outlet to 31.5" OD Piping Inlet	30" OD / 0.9742" min wall	28.0516	8,572	142.9	8,835	147.2	8,657	144.3	8,487	141.4	8,966	149.4	8,786	146.4	8,612	143.5
31.5" OD Piping Inlet to 31.5"OD Pipign Outlet	31" OD / 1.461" min wall	28.078	8,648	144.1	8,920	148.7	8,738	145.6	8,562	142.7	9,056	150.9	8,870	147.8	8,691	144.9
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,685	144.7	8,959	149.3	8,775	146.3	8,598	143.3	9,096	151.6	8,909	148.5	8,729	145.5
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,686	144.8	8,960	149.3	8,776	146.3	8,599	143.3	9,097	151.6	8,910	148.5	8,729	145.5
30" x 24" Cross to Steam Dump Branch (18" Line to Atmosphere)	30" OD / 0.9742" min wall	28.0516	8,696	144.9	8,972	149.5	8,787	146.5	8,609	143.5	9,109	151.8	8,921	148.7	8,740	145.7
Steam Dump Branch (18" Line to Atmosphere) to Steam Dump Branch (18" Line to Condenser)	30" OD / 0.9742" min wall	28.0516	8,786	146.4	9,071	151.2	8,881	148.0	8,698	145.0	9,213	153.6	9,019	150.3	8,833	147.2
From SG A Main Header to Takeoff to FCV484B	18" OD / Sch 80	16.124	10,803	180.0	11,166	186.1	10,924	182.1	10,693	178.2	11,346	189.1	11,100	185.0	10,864	181.1
Takeoff to FCV484B to 6" MSR Takeoff	18" OD / Sch 80	16.124	7,355	122.6	7,614	126.9	7,443	124.0	7,279	121.3	7,742	129.0	7,568	126.1	7,401	123.3
6" MSR Takeoff to Takeoff to FCV484C	18" OD / Sch 80	16.124	7,362	122.7	7,622	127.0	7,450	124.2	7,286	121.4	7,751	129.2	7,576	126.3	7,408	123.5
Takeoff to FCV484C to Takeoff to FCV484A	18" OD / Sch 80	16.124	3,685	61.4	3,815	63.6	3,729	62.1	3,647	60.8	3,880	64.7	3,792	63.2	3,708	61.8
8" Takeoff to FCV484A	8" OD / Sch 80	7.625	16,480	274.7	17,064	284.4	16,678	278.0	16,310	271.8	17,354	289.2	16,960	282.7	16,584	276.4
8" Takeoff to FCV484B	8" OD / Sch 80	7.625	16,443	274.1	17,023	283.7	16,640	277.3	16,274	271.2	17,311	288.5	16,920	282.0	16,546	275.8
8" Takeoff to FCV484C	8" OD / Sch 80	7.625	16,476	274.6	17,060	284.3	16,674	277.9	16,306	271.8	17,349	289.2	16,956	282.6	16,580	276.3

TABLE 7B

Condenser Dump Valve Piping Velocities
Steam Generator B

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Steam Generator Pressure, psia			740.50		767.00		782.00		797.00		767.00		782.00		797.00	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator B Outlet to 31.5" OD Piping	30" OD / 0.9742" min wall	28.0516	8,572	142.9	8,835	147.2	8,657	144.3	8,487	141.4	8,966	149.4	8,786	146.4	8,612	143.5
31.5" OD Piping Inlet to 31.5"OD Piplgn Outlet	31" OD / 1.461" min wall	28.078	8,649	144.2	8,922	148.7	8,739	145.6	8,563	142.7	9,057	150.9	8,871	147.9	8,693	144.9
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	8,687	144.8	8,961	149.4	8,777	146.3	8,600	143.3	9,098	151.6	8,911	148.5	8,730	145.5
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	8,688	144.8	8,963	149.4	8,778	146.3	8,601	143.3	9,099	151.7	8,912	148.5	8,731	145.5
30" x 24" Cross to Steam Dump / MSR Branch (18" Stm Dump to Condenser)	30" OD / 0.9742" min wall	28.0516	8,698	145.0	8,974	149.6	8,789	146.5	8,611	143.5	9,111	151.9	8,923	148.7	8,743	145.7
From SG B Main Header to Takeoff to FCV484F	18" OD / Sch 80	16.124	10,641	177.4	10,987	183.1	10,756	179.3	10,534	175.6	11,159	186.0	10,924	182.1	10,698	178.3
Takeoff to FCV484F to 6" MSR Takeoff	18" OD / Sch 80	16.124	7,204	120.1	7,446	124.1	7,285	121.4	7,130	118.8	7,566	126.1	7,402	123.4	7,245	120.7
6" MSR Takeoff to Takeoff to FCV484E	18" OD / Sch 80	16.124	7,211	120.2	7,454	124.2	7,292	121.5	7,137	119.0	7,574	126.2	7,410	123.5	7,252	120.9
Takeoff to FCV484E to Takeoff to FCV484D	18" OD / Sch 80	16.124	3,609	60.1	3,731	62.2	3,650	60.8	3,572	59.5	3,791	63.2	3,709	61.8	3,630	60.5
8" Takeoff to FCV484D	8" OD / Sch 80	7.625	16,141	269.0	16,687	278.1	16,324	272.1	15,976	266.3	16,958	282.6	16,588	276.5	16,234	270.6
8" Takeoff to FCV484E	8" OD / Sch 80	7.625	16,137	269.0	16,683	278.0	16,320	272.0	15,973	266.2	16,953	282.6	16,584	276.4	16,230	270.5
8" Takeoff to FCV484F	8" OD / Sch 80	7.625	16,106	268.4	16,648	277.5	16,288	271.5	15,942	265.7	16,917	282.0	16,550	275.8	16,198	270.0

TABLE 8A

SRV Piping Velcocities
Steam Generator A

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		. 1780		1780		1780	
Calculated Pressure @ SG A, psia			1177.3		1182.3		1182.3		1201.6		1203.2		1203.3		1203.3	
Piping Description	**Pipe Size/Schedule**	**d inches**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**	**Velocity ft / min**	**Velocity ft / sec**
Steam Generator A Outlet to 31.5" OD Piping Inlet	30" OD / 0.9742" min wall	28.0516	5,142	85.7	5,474	91.2	5,476	91.3	5,376	89.6	5,443	90.7	5,445	90.8	5,447	90.8
31.5" OD Piping Inlet to 31.5"OD Pipign Outlet	31" OD / 1.461" min wall	28.078	5,155	85.9	5,491	91.5	5,493	91.6	5,392	89.9	5,460	91.0	5,462	91.0	5,464	91.1
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	5,169	86.2	5,508	91.8	5,509	91.8	5,408	90.1	5,476	91.3	5,478	91.3	5,480	91.3
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	5,170	86.2	5,508	91.8	5,510	91.8	5,408	90.1	5,477	91.3	5,478	91.3	5,480	91.3
SG A 30" x 24" Cross to SD23-2 Takeoff	24" OD / Sch 80	21.562	3,502	58.4	3,731	62.2	3,732	62.2	3,664	61.1	3,710	61.8	3,711	61.9	3,712	61.9
SD23-2 Takeoff to SD23-1 Takeoff	24" OD / Sch 80	21.562	1,752	29.2	1,867	31.1	1,868	31.1	1,833	30.6	1,856	30.9	1,857	31.0	1,858	31.0
SD23-2 Takeoff to SD23-2	6" OD / Sch 80	5.761	24,543	409.0	26,153	435.9	26,162	436.0	25,679	428.0	26,004	433.4	26,013	433.6	26,022	433.7
SD23-1 Takeoff to SD23-1	6" OD / Sch 80	5.761	24,544	409.1	26,155	435.9	26,164	436.1	25,681	428.0	26,006	433.4	26,015	433.6	26,024	433.7
SG A 30" x 24" Cross to SD23-3 Takeoff	24" OD / Sch 80	21.562	5,252	87.5	5,596	93.3	5,598	93.3	5,495	91.6	5,565	92.7	5,567	92.8	5,569	92.8
SD23-3 Takeoff to SD23-4 Takeoff	24" OD / Sch 80	21.562	3,507	58.5	3,738	62.3	3,739	62.3	3,670	61.2	3,716	61.9	3,718	62.0	3,719	62.0
SD23-4 Takeoff to SD23-5 Takeoff	24" OD / Sch 80	21.562	1,754	29.2	1,869	31.2	1,870	31.2	1,835	30.6	1,859	31.0	1,859	31.0	1,860	31.0
SD23-3 Takeoff to SD23-3	6" OD / Sch 80	5.761	24,564	409.4	26,179	436.3	26,188	436.5	25,704	428.4	26,030	433.8	26,039	434.0	26,048	434.1
SD23-4 Takeoff to SD23-4	6" OD / Sch 80	5.761	24,570	409.5	26,186	436.4	26,195	436.6	25,711	428.5	26,037	434.0	26,046	434.1	26,055	434.3
SD23-5 Takeoff to SD23-5	6" OD / Sch 80	5.761	24,571	409.5	26,188	436.5	26,197	436.6	25,713	428.5	26,039	434.0	26,048	434.1	26,057	434.3

TABLE 8B

SRV Piping Velocities
Steam Generator B

Heat Balance Description			HB101		HB102		HB103		HB104		HB105		HB106		HB107	
Steam Generator Power, MWt			1656		1757		1757		1757		1780		1780		1780	
Calculated Pressure @ SG B, psia			1,177.4		1,182.4		1,182.4		1,201.7		1,203.3		1,203.4		1,203.4	
Piping Description	Pipe Size/Schedule	d Inches	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec	Velocity ft / min	Velocity ft / sec
Steam Generator B Outlet to 31.5" OD Piping	30" OD / 0.9742" min wall	28.0516	5,141	85.7	5,474	91.2	5,475	91.3	5,376	89.6	5,443	90.7	5,444	90.7	5,446	90.8
31.5" OD Piping Inlet to 31.5"OD Pipign Outlet	31" OD / 1.461" min wall	28.078	5,155	85.9	5,491	91.5	5,493	91.5	5,392	89.9	5,460	91.0	5,462	91.0	5,464	91.1
31.5" OD Piping Outlet to TD AFW Pump Branch	30" OD / 0.9742" min wall	28.0516	5,169	86.2	5,508	91.8	5,509	91.8	5,408	90.1	5,476	91.3	5,478	91.3	5,480	91.3
TD AFW Pump Branch to 30" x 24" Cross	30" OD / 0.9742" min wall	28.0516	5,170	86.2	5,508	91.8	5,510	91.8	5,408	90.1	5,477	91.3	5,478	91.3	5,480	91.3
SG B 30" x 24" Cross to SD23-9 Takeoff	24" OD / Sch 80	21.562	3,502	1,171.3	3,731	1,175.5	3,732	1,175.5	3,664	1,194.9	3,710	1,196.3	3,711	1,196.4	3,712	1,196.4
SD23-9 Takeoff to SD23-10 Takeoff	24" OD / Sch 80	21.562	1,752	1,170.6	1,867	1,174.7	1,868	1,174.7	1,833	1,194.1	1,856	1,195.5	1,857	1,195.6	1,858	1,195.6
SD23-9 Takeoff to SD23-9	6" OD / Sch 80	5.761	24,543	1,170.6	26,153	1,174.7	26,162	1,174.7	25,679	1,194.1	26,004	1,195.5	26,013	1,195.6	26,022	1,195.6
SD23-10 Takeoff to SD23-10	6" OD / Sch 80	5.761	24,544	1,170.5	26,155	1,174.6	26,164	1,174.6	25,681	1,194.0	26,006	1,195.4	26,015	1,195.5	26,024	1,195.5
SG A 30" x 24" Cross to SD23-8 Takeoff	24" OD / Sch 80	21.562	5,252	1,171.3	5,596	1,175.5	5,598	1,175.5	5,495	1,194.9	5,565	1,196.3	5,567	1,196.4	5,569	1,196.4
SD23-8 Takeoff to SD23-7 Takeoff	24" OD / Sch 80	21.562	3,507	1,169.7	3,738	1,173.7	3,739	1,173.7	3,670	1,193.1	3,716	1,194.5	3,718	1,194.5	3,719	1,194.6
SD23-7 Takeoff to SD23-6 Takeoff	24" OD / Sch 80	21.562	1,754	1,169.5	1,869	1,173.4	1,870	1,173.4	1,835	1,192.8	1,859	1,194.2	1,859	1,194.3	1,860	1,194.3
SD23-8 Takeoff to SD23-8	6" OD / Sch 80	5.761	24,564	1,169.7	26,179	1,173.7	26,188	1,173.7	25,704	1,193.1	26,030	1,194.5	26,039	1,194.5	26,048	1,194.6
SD23-7 Takeoff to SD23-7	6" OD / Sch 80	5.761	24,570	1,169.5	26,186	1,173.4	26,195	1,173.4	25,711	1,192.8	26,037	1,194.2	26,046	1,194.3	26,055	1,194.3
SD23-6 Takeoff to SD23-6	6" OD / Sch 80	5.761	24,571	1,169.4	26,188	1,173.3	26,197	1,173.3	25,713	1,192.8	26,039	1,194.1	26,048	1,194.2	26,057	1,194.2

ENCLOSURE 4

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

January 30, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

NMC Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195

REGULATORY COMMITMENTS UPDATE

3 Pages Follow

LIST OF REGULATORY COMMITMENTS (Update)

The following table identifies those actions committed to by NMC made in association with LAR 195. Any other statements in this submittal or prior submittals associated with LAR 195 are provided for information purposes and are not considered to be regulatory commitments.

The purpose of restating these commitments is to provide the NRC staff with an up-to-date status of commitment completion and those for which actions are still required. The changes to this table are that commitments 9 and 11 have been completed.

	Commitment Description	Due Date/Event
1	Section 7.1.2 of attachment 4 states that an evaluation of the thermal and hydraulic safety analyses for the Framatome ANP fuel demonstrates that the DNBR design basis is met for the Framatome ANP fuel in Cycle 26. This evaluation has been performed for cycle 26 reload safety evaluation, the COLR, and the associated 10 CFR 50.59. All documents will be revised, as appropriate, for the stretch power uprate to address Framatome fuel DNBR design basis.	Prior to implementation of the stretch power uprate.
2	Sections 8.3.3.3 and 8.3.7.3 of attachment 4 state the increase in flow rate and velocities, as well as the changes in operating pressures and temperatures, will be incorporated into the KNPP FAC program as part of the power uprate implementation. The FAC program models will be updated prior to the next program inspections scheduled for the next refueling outage.	Prior to the next scheduled refueling outage.
3	Section 8.3.3.3, 8.3.15, 8.6.4, 8.6.5, and 8.6.6 of attachment 4 all refer to the feedwater control valve trim modification. This modification was completed during the R26 refueling outage.	Complete.
4	Section 8.3.3.3 of attachment 4 describes the following in regards to the feedwater heaters: a. Initial inspection and analyses will establish a baseline prior to the stretch uprate implementation. These baseline inspections and analyses were completed during the R26 refueling outage. b. An inspection and monitoring program will be established to monitor potential heater degradation at the stretch power uprate conditions. An inspection program will be developed based on the baseline inspection results and using programs and processes in place at KNPP. This will be completed prior to the next refueling outage.	a. Complete. b. Prior to the next refueling outage.

	Commitment Description	Due Date/Event
5	Sections 8.3.8.4 and 8.4.2.2.1 of attachment 4 state that a cycle-specific heat load calculation will be performed prior to each refueling outage to determine the adequacy of spent fuel pool cooling capability. Reactor engineering refueling procedures have already incorporated this confirmatory calculation as a requirement. The new requirement will administratively control the in core hold time of the fuel after shutdown to ensure the requirements are met.	Complete.
6	Instrument and Control System Setpoint changes were summarized in sections 8.3.9.3 and section 8.3.15 of attachment 4. These recommended changes included CST level setpoints, first stage turbine pressure, and turbine overspeed trip settings. The turbine overspeed trip setting changes were completed during the R26 refueling outage. The other recommendations will be reviewed by the plant staff and implemented as appropriate.	Prior to stretch uprate implementation.
7	Piping and pipe support evaluations concluded in section 8.4.4 that the systems remain acceptable assuming resolution of open items. Open items remained on the following systems: service water and component cooling water. These open items will be resolved.	Prior to stretch uprate implementation.
8	Modifications to the steam generator level control system were recommended to support the stretch uprate as described in Sections 8.6.5 and 8.6.6 of attachment 4. The level control changes were completed during the R26 refueling outage.	Complete.
9	Evaluation of the EQ equipment inside containment affected by the higher containment EQ long-term temperature profile will be performed per Sections 8.9.3 and 8.9.4 of attachment 4.	Complete.
10	The KNPP EQ Plan will be updated, as appropriate, to reflect power uprate evaluations in accordance with Section 8.8.4.3 of attachment 4.	Prior to implementation of stretch power uprate.
11	For those components where the HELB temperatures exceeded the equipment qualification temperature, the EQ equipment required for HELB outside containment will be qualified per Section 8.9.3 and 8.9.4 of attachment 4.	Complete.
12	Plant procedures will be revised as appropriate to accommodate the stretch power uprate. Procedure changes were committed to in Attachment 1, Section 2.6 and in Attachment 4, Section 8.7.2. Emergency, abnormal, and operating procedures that are entered due to a LONF event or have AFW TS requirements will be changed as appropriate to reflect the new TS requirements for the AFW system.	Prior to implementation of stretch power uprate.

	Commitment Description	Due Date/Event
13	New TS requirements, revised procedures, and any control room changes due to the stretch power uprate will be reviewed by training for determination of being included in the operator training program. This was committed to in Attachment 1, Section 2.6.	Prior to implementation of stretch power uprate.
14	Attachment 1, Section 2.6 stated that setpoint changes for reactor protection and control inputs, alarms, computer constants, and embedded values, will be updated consistent with operation at 1772 MWt. Power range nuclear instruments will be recalibrated and checked based on a secondary heat balance.	Prior to implementation of stretch power uprate
15	Sections 9.2.4.1.7 and 9.2.4.2.5 of attachment 4 describe recommended modifications to the high pressure turbine cylinder joint bolting and the low-pressure turbine coupling bolts. These modifications were completed during the recent R26 refueling outage.	Complete.
16	The response to RAI question #57 to the stretch power uprate submittal stated degraded voltage and thrust calculations for MOV operators outside containment were reviewed for impact of the uprated post accident temperatures, and will be revised, as required, prior to implementation of the 6% Stretch Power Uprate.	Prior to implementation of stretch power uprate.
17	NMC will provide NRC with a status update and schedule for resolution of GL 96-06 water hammer issues	April 2, 2004

Note: attachment 4 refers to attachment 4 of reference letter 1 listed on the cover letter.

ENCLOSURE 5

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

January 30, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

NMC Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195

NMC Responses to Letter from John Lamb (NRC) to Thomas Coutu (NMC), "Kewaunee Nuclear Power Plant - Review of License Amendment Request No. 195, Stretch Power Uprate (TAC NO. MB9031)," dated January 26, 2004)

2 Pages Follow

In the referenced letter (reference 4) from the NRC, the NRC stated that it still needed additional technical information to complete its review. Below is a listing of the information the staff stated it needed and NMC's response or direction to where NMC's response is located in enclosure 2 of this letter:

(1) an evaluation to address the effects of the stretch power uprate on utilizing the refueling water storage tank to borate the reactor coolant system to cold shutdown, with and without the availability of letdown,

NMC Response: See NMC response to NRC SPLB Branch question number 2 found on page 10 of enclosure 2.

(2) the calculations and description of the methodology to eliminate the rod bow penalty,

NMC Response: See NMC response to NRC SRXB Branch question number 2 found on page 22 of enclosure 2.

(3) the technical basis for operating the feedwater and condensate motors at service factors of approximately 1.03 at stretch power uprate conditions,

NMC Response: See NMC response to NRC EMEB Branch question number 5 found on page 4 of enclosure 2.

(4) the technical basis for operating at the stretch power uprate conditions in light of the water hammer, two-phase flow, and thermal overpressurization concerns that are discussed in Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During a Design-Basis Accident Conditions," dated September 30,1996,

NMC Response: See NMC response to NRC SPLB Branch question number 5 found on page 19 of enclosure 2.

(5) a summary of your responses to post-Three Mile Island (TMI) Action Plan Item II.E.LI, that includes a listing of applicable correspondence and a description of the impact that the proposed stretch power uprate will have on your responses to and resolution of this particular post-TMI Action Plan item,

NMC Response: See NMC response to NRC SPLB Branch question number 4 found on page 14 of enclosure 2.

(6) how you interpret the proposed auxiliary feedwater (AFW) TS requirements if a steam supply to the turbine-driven AFW pump is inoperable when another AFW pump is inoperable,

NMC Response: Based on current KNPP TS 3.4.b.2, when a steam supply to the Turbine Driven AFW (TDAFW) pump is inoperable a 7-day allowed outage time (AOT) limitation is entered (KNPP TS 3.4.b.2.C). As stated in the KNPP TS basis, this condition does not render the TDAFW pump inoperable or render an AFW train inoperable (KNPP TS basis page TS B3.4-1). If one of the AFW pumps is concurrently declared inoperable then one train of AFW is inoperable and a 72-hour AOT is entered (KNPP TS 3.4.b.2.A). The NRC in the staffs' safety evaluation for KNPP license amendment 123 approved this condition.[1]

(7) how you plan to coordinate the down power maneuver when two AFW pumps are inoperable to assure decay heat removal requirements will be met in the event of a loss of normal feedwater event, and

NMC Response: With the proposed stretch power uprate TS, during 100% power operation (1772 Mwth) when two AFW pumps are inoperable (i.e. two AFW trains are inoperable) two limiting conditions for operation (LCO's) are entered simultaneously. The first is proposed TS 3.4.b.3, "If two of the three AFW trains are inoperable, then within two hours, reduce reactor power to ≤ 1673 MWth. The second is proposed TS 3.4.b.4.B (current TS 3.4.b.2.B), "Two auxiliary feedwater trains may be inoperable for 4 hours. Thus within two hours of the time of discovery of two AFW trains inoperable the plant power will be reduced to ≤ 1673 MWth, if the condition is not corrected within 4 hours of the time of discovery the plant will shutdown in accordance with proposed TS 3.4.b.6 with a final end state of the reactor coolant system temperature < 350°F

(8) an example showing that sufficient margin exists between the analytical limit and the setpoint including the tolerance band around the setpoint to illustrate when the channel is considered inoperable.

NMC Response: See NMC response to NRC EEIB Branch question number 1 found on page 24 of enclosure 2.

Enclosure 5 References:

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[1] Letter from Richard J. Laufer (NRC) to M.L. Marchi (WPSC), "Amendment No. 123 To Facility Operating License No. DPR-43 – Kewaunee Nuclear Power Plant (TAC NO. M93977), dated January 3, 1996,