February 9, 2004

NRC-04-015
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.
5) 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 (TAC NO. MB9031) dated January 30, 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.

On November 5, 2003; December 15, 2003; and January 30, 2004; NMC responded to requests for additional information from the Nuclear Regulatory Commission (NRC) regarding the proposed stretch power uprate (References 2, 3, and 5). Subsequent to NMC's January 30th response, the NRC staff has requested clarification on some of the responses submitted in NMC's January 30th letter. This letter and enclosures contain the NMC responses to the NRC requests for clarification.

Enclosure 1 contains the clarification questions from the NRC. These questions were derived from emails received by NMC from the NRC staff. Enclosure 2 contains NMC's responses to these clarification questions.

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.

<u>Summary of Commitments</u>

This letter makes no new commitments.

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 February 9, 2004.

Kevin Davison for

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

GOR

Enclosures: (2)

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

February 9, 2004

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

<u>Responses to NRC Clarification Questions on Responses to Requests for Additional Information Regarding LAR 195</u>

NRC Clarification Questions on Responses to Requests for Additional Information

1 Page Follows

NRC Question 1:

Regarding the boron dilution event, did the licensee evaluate this event for the Framatome ANP fuel, or is this bounded by the analysis performed by Westinghouse for the 422V+ fuel? I'm thinking of the transition core period where both fuel types will exist, for example, Cycle 26.

NRC Question 2:

Regarding the rod ejection event, does the Westinghouse analysis for the 422 V+ fuel bound the Framatome fuel analysis?

NRC Question 3:

The 1/30/2004 letter states that the boric acid solubility limit of 38 w/o% used is for saturated conditions at 20 psig with a 4 w/o% margin. This is compared to the value of 23.53 w/o% at one atmosphere, again with a 4 w/o% margin. The difference between the Kewaunee-predicted 18 hours and the 6 - 9 hours for non-UPI plants (see ML03270538) is attributed to the difference between these assumed pressure conditions. No information is provided to substantiate this claim. The staff requires a Kewaunee-specific analysis that establishes the time at which a saturated condition will be reached assuming a one atmosphere pressure. The description of this analysis shall fully describe all analysis assumptions and the calculation process.

The 1/30/2004 letter states that "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." The staff requires substantiation for the "expectation" that effective upper plenum injection will be initiated and maintained prior to the boric acid concentration reaching the saturation condition that would exist at a pressure of one atmosphere. (An alternative to this staff requirement would be to conservatively establish that boric acid precipitation will not occur during cooldown following a condition in which the boric acid concentration exceeds the one atmosphere saturation concentration.) The substantiation required by the staff shall include a description of the applicable procedures, the timing of the applicable principal steps in the procedures, the equipment required for the cooldown/depressurization (including the equipment safety-related "pedigree"), and substantiation of the timing (such as operating, training, and simulator experience).

ENCLOSURE 2

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

February 9, 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

7 Pages Follow

NRC Question 1:

Regarding the boron dilution event, did the licensee evaluate this event for the Framatome ANP fuel, or is this bounded by the analysis performed by Westinghouse for the 422V+ fuel? I'm thinking of the transition core period where both fuel types will exist, for example, Cycle 26.

NMC Response:

The boron concentrations (initial and critical) assumed in the boron dilution analyses account for the mixed core. The calculated reactor vessel volume, which is part of the active RCS volume, corresponds to a full core of Westinghouse 422 V+ fuel. However, since the Framatome and Westinghouse fuel rod dimensions and fuel assembly water volumes are nearly identical, the impact of mixed core on the calculated operator action times would be negligible.

The the boron dilution transient was evaluated for the Framatome ANP fuel. The results of this evaluation were acceptable Thus; the boron dilution analysis results are applicable to a mixed core (Westinghouse and Framatome ANP fuel), for example Cycle 26.

NRC Question 2:

Regarding the rod ejection event, does the Westinghouse analysis for the 422 V+ fuel bound the Framatome fuel analysis?

NMC Response:

Both fuel types were explicitly analyzed in the rod ejection analysis, and the Westinghouse 422V+ results were bounding for the Framatome ANP fuel results.

NRC Question 3:

The 1/30/2004 letter states that the boric acid solubility limit of 38 w/o% used is for saturated conditions at 20 psig with a 4 w/o% margin. This is compared to the value of 23.53 w/o% at one atmosphere, again with a 4 w/o% margin. The difference between the Kewaunee-predicted 18 hours and the 6 - 9 hours for non-UPI plants (see ML03270538) is attributed to the difference between these assumed pressure conditions. No information is provided to substantiate this claim. The staff requires a Kewaunee-specific analysis that establishes the time at which a saturated condition will be reached assuming a one atmosphere pressure. The description of this analysis shall fully describe all analysis assumptions and the calculation process.

The 1/30/2004 letter states that "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." The staff requires substantiation for the "expectation" that effective upper plenum injection will be initiated and maintained prior to the boric acid concentration reaching the saturation condition that would exist at a pressure of one atmosphere. (An alternative to this staff requirement would be to conservatively establish that boric acid precipitation will not occur during cooldown following a condition in which the boric acid concentration exceeds the one atmosphere saturation concentration.) The substantiation required by the staff shall include a description of the applicable procedures, the timing of the applicable principal steps in the procedures, the equipment required for the cooldown/depressurization (including the equipment safety-related "pedigree"), and substantiation of the timing (such as operating, training, and simulator experience).

NMC Response:

As part of the NRC review of the Kewaunee Nuclear Power Plant (KNPP) Power Uprate (PU) Program a number of RAIs have been addressed regarding the potential for boric acid precipitation in the core after a loss of coolant accident (LOCA). In response to the new specific NRC RAI above regarding the potential for boric acid precipitation at atmospheric conditions after a LOCA for the KNPP PU program, existing KNPP PU boric acid precipitation calculations were reviewed.

The results of this review indicated that at atmospheric conditions, the boric acid concentration in the core region would remain 4 wt. % below the established boron precipitation limit for at least 7.8 hours after the LOCA.

The existing calculations were based on the methodology and assumptions listed below. These assumptions are consistent with, or otherwise conservative with respect to, the assumptions described in Reference 1. Outside of the assumption of atmospheric pressure, the methodology and assumptions listed below are also consistent with the post LOCA boron precipitation analyses that were part of the KNPP replacement steam generator (RSG) Program and the reload transition safety report (RTSR) Program.

- A boric acid concentration level is computed over time following a LOCA for a core-region mixing volume. Other than the steam exiting through the reactor coolant system (RCS) hot legs and the corresponding makeup safety injection (SI) entering through the reactor vessel lower plenum, there are no other assumed flow paths in or out of the core-region mixing volume. All boric acid entering the mixing volume remains in the mixing volume prior to initiation of RCS hot leg recirculation (or for KNPP, prior to upper plenum injection). The water/boric acid solution is well mixed in the mixing volume region. The water/boric acid solution in the reactor vessel is assumed to be at atmospheric conditions, at a temperature of 212°F. The collapsed mixture level of the core/upper plenum region is at the bottom of the RCS hot leg flow area at the reactor vessel outlet nozzle. This level is the top of the mixing volume. The bottom of the mixing volume is at the level of the top of the lower core support plate. The reactor vessel lower plenum volume and barrel baffle region volume are not included in the mixing volume.

- The boric acid concentration precipitation limit is the experimentally determined boric acid saturation concentration with a four weight-percent uncertainty factor. This boron precipitation limit is established to be 23.53 wt. %. There is no allowance for increase in boric acid solubility due to other solutes such as sodium hydroxide. The boron precipitation limit calculation neglects any elevation of boiling temperature due to concentration of boric acid in the core or due to backpressure from containment.
- The decay heat generation rate is based on the 1971 ANS Standard for a finite operating time. The decay heat generation rate includes a core power multiplier to address instrumentation uncertainty as identified by Section I.A of 10CFR 50 Appendix K.
- The boron concentration of the make-up SI is a calculated containment sump mixed mean boron concentration. The calculation of the containment sump mixed mean boron concentration assumes maximum mass and maximum boron concentrations for significant boron sources and minimum mass and maximum boron concentration for significant dilution sources.

Following is a summary of the operator actions including times for the significant actions that are required for mitigation of a small break loss of coolant accident (SBLOCA). These actions and times are based on a detailed human reliability analysis that was performed using the KNPP emergency operating procedures, KNPP human reliability experience, and KNPP operator measured response data.

Time Line to Establish RHR Head Injection			
Elapsed time (Minutes)	Procedure Step	Basis	Comments
Initial Conditions and Assumptions: • Only one train of Safety Injection is available for the event. • Cooldown rate is limited to 50 DEG F/hr although the allowed rate is 100 DEG F/hr. • Both Steam Generators and Trains of RHR are available for cooldown. • The break is assumed to be approximately 1.25 inches in diameter			
0		**LOCA Event Occurs**	
0		LOCA size needs to be less than 2 inches and greater than 1.1 inches to be a concern for boron precipitation. If LOCA break area is too large, RHR injection will begin automatically or with minimal operator response. IF LOCA break area is too small, safety injection flow will overpower the mass loss.	MAAP run 125 LOCA provides a reference for the size of LOCA.
1	E-0, "Reactor Trip or Safety Injection" step 1	Simulator observation on 11/4/02 of Crew A (during weekly dynamic scenario).	Operators respond and enter E-0 High Head Safety Injection is QA-01, Safety Related System, Structure, Component (SSC).
18	E-0, steps 1-23		Operators complete the actions of E-0 and transition to E-1.
39	E-1, "Loss of Reactor or Secondary Coolant" steps 1-18		Operators perform the actions of E-1 steps 1-18 and transition to ES-1.2.

Time Line to Establish RHR Head Injection			
Elapsed time (Minutes)	Procedure Step	Basis	Comments
44	ES-1.2, "Post LOCA Cooldown and Depressurization" step 1 ES-1.2 step 5	Simulator observation and based on operator interviews conducted from 9/25/02 to 6/3/03, it was determined the average time to perform a control room operator action was 0.8 minutes. This time does not include plant response time for the operator-performed action such as RCS temperature response after initiation of a cooldown.	
<u>60</u>		<u>**Cool down Begins**</u>	<u>Assumed time = 60 minutes</u>
70	ES-1.2 steps 6-23	Simulator observation and based on operator interviews conducted from 9/25/02 to 6/3/03, it was determined the average time to perform a control room operator action was 0.8 minutes. This time does not include plant response time for the operator-performed action such as RCS temperature response after initiation of a cooldown.	RCS Cooldown using Steam Generator PORV's. SG PORV's are QA-2 Components. Steam Generators and the Auxiliary Feedwater System are QA-1, Safety Related SSC's.
120	ES-1.2 step 24	MAAP run 125 LOCA showed RHR conditions occur in 2 hours. Time to RHR conditions will vary with the cooldown rate and leak size. Thus time to reach RHR conditions is conservatively assumed to be 4 hours.	It is assumed that the Emergency Director would direct RHR Split Train Mode per A-RHR-34B and the operators would take action to reach conditions to place RHR and low head SI in service.

Time Line to Establish RHR Head Injection			
Elapsed time (Minutes)	Procedure Step	Basis	Comments
240		**RHR (LHSI) Alignment Begins**	Assumed time = 240 minutes
	A-RHR-34B, "Residual Heat Removal Split-Train Mode" Step 4.1-4.4	Simulator observation and based on operator interviews conducted from 9/25/02 to 6/3/03, it was determined the average time to perform a control room operator action was 0.8 minutes. This time does not include plant response time for the operator-performed action such as RCS temperature response after initiation of a cooldown. Times are based on an interview of an operator conducted 1/24/03 for N-RHR-34, "Residual Heat Removal System Operation", for the time to align the RHR system per A-RHR-34B.	RHR (LHSI) is a Safety Related, QA-1 SSC.
396		Assuming the 50 DEG F /hr cooldown continues during the alignment of the RHR system, RCS temperature would be 300 DEG F.	It is assume RCS pressure will be reduced to <150 psig and establish Upper Head Injection as soon as possible. For the injection to occur, the operator would only be required to place RHR Pump B to start. 300 DEG F corresponds to a saturation pressure of 67 psia. Allowing for 30 DEG F sub-cooling, 106 psia is the required RCS pressure thus Upper Head Injection can be established at this time.
396		**RHR is established with LHSI Upper Plenum Injection**	Conservative time for LHSI to Upper Plenum = 396 minutes.

KNPP Emergency Procedures Used	Description
E-0	Reactor Trip or Safety Injection
E-1	Loss of Reactor or Secondary Coolant
ES-1.2	Post LOCA Cooldown and Depressurization
A-RHR-34B	Residual Heat Removal Split-Train Mode
N-RHR-34	Residual Heat Removal System Operation

Therefore, based on a conservative analysis of operator actions and action times using KNPP emergency operating procedures and human reliability data/experience, upper plenum injection of emergency core cooling system (ECCS) low head safety injection (LHSI) water directly into the reactor vessel above the reactor core is established at 6.6 hours. Since the time to establish LHSI to the reactor upper plenum (6.6 hours) is less than the time to reach the boron precipitation limit at one atmosphere (7.8 hours) there is no potential for boron precipitation following the limiting design basis LOCA accidents.

References:

1. Letter from C. L. Caso to T. M. Novak, Chief, Reactor Systems Branch, NRC, from Manager, Safeguards Engineering, Westinghouse Corporation Power Systems, CLC-NS-309, April 1, 1975.