January 23, 2008

Mr. Stewart B. Minahan
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD7385)

Dear Mr. Minahan:

By application dated November 19, 2007, Nebraska Public Power District (NPPD, the licensee) requested the U.S. Nuclear Regulatory Commission (NRC) staff approval of an amendment to the Cooper Nuclear Station facility operating license and technical specifications to increase the rated thermal power from 2381 to 2419 megawatts thermal (1.62 percent increase) based upon increased feedwater flow measurement accuracy to be achieved by using high accuracy ultrasonic flow measurement instrumentation. Your proposal is a measurement uncertainty recapture power uprate.

The NRC staff has reviewed the information provided in your submittal and determined that additional information is required in order to complete its review. Please provide a response to the enclosed questions by March 3, 2008.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-2296.

Sincerely,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: Request for Additional Information

cc w/encl: See next page

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO MEASUREMENT UNCERTAINTY RECAPTURE

POWER UPRATE REQUEST

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

I. The following questions are provided from the Steam Generator and Chemical Engineering Branch (CSGB):

1. The flow accelerated corrosion (FAC) monitoring program includes the use of a predictive method to calculate the wall thinning of components susceptible to FAC. In order for the U.S. Nuclear Regulatory Commission (NRC) staff to evaluate the accuracy of these predictions, the staff requests a sample list of components for which wall thinning is predicted and measured by ultrasonic testing or other methods. Include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the measured wall thickness to the thickness predicted by the model.

2. The power uprate will affect several process variables that influence FAC. Identify the systems that are expected to experience the greatest increase in wear as a result of the power uprate and discuss the effect of individual process variables (i.e., moisture content, temperature, oxygen, and flow velocity) on each system identified. For the most susceptible systems and components, what is the total predicted increase in wear rate due to FAC as a result of power uprate conditions?

II. The following question is provided from the Vessels and Internals Integrity Branch (CVIB):

1. Table 3-1 of Enclosure 1 to the submittal reported the peak end-of-license, i.e., 32 effective full power years (EFPY), reactor vessel (RV) inside diameter (ID) fluence, considering the measurement uncertainty recapture (MUR) power uprate, as 1.68×10^{18} n/cm^2 (E>1.0 MeV) for lower-intermediate shell plates and all welds. Based on this, the staff estimated that the ID fluence for 30 EFPY is 1.575×10^{18} n/cm^2 (E>1.0 MeV) [1.68x30/32]. The current Cooper Nuclear Station (CNS) technical specifications contain pressure-temperature (P-T) limit curves valid for 30 EFPY based on a projected peak RV ID fluence of 1.57×10^{18} n/cm^2 (E>1.0 MeV) for the limiting beltline material (the lower-intermediate longitudinal weld), as evaluated in the safety evaluation dated January 24, 2006. Please explain the need to revise the CNS P-T limit curves to 28 EFPY.

III. The following questions are provided from the Fire Protection Branch (AFPB):

1. The staff notes that the General Electric Company (GE)-Hitachi Safety Analysis Report for CNS Thermal Power Optimization (TPO), NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that operation of the plant at the TPO level does not affect fire detection and suppression systems. Please address the impact of TPO uprate conditions on other fire protection program elements. At a minimum, include the following: (1) administrative controls, (2) fire barriers, (3) fire protection responsibilities of plant personnel, and (4) procedures and resources necessary for systems required to achieve and maintain safe-shutdown.

2. The staff notes that the GE-Hitachi Safety Analysis Report for CNS TPO, NEDC-33385P, Revision 0, November 2007, Section 6.7, "Fire Protection," states that the operator actions required to mitigate the consequences of a fire are not affected. Please verify that additional heat in the plant environment (from the MUR power uprate) will not interfere with required operator manual actions being performed at their designated time.

3. The results of the Appendix R evaluation for the MUR power uprate are provided Section 6.7, "Fire Protection," of the GE-Hitachi Safety Analysis Report for CNS TPO, NEDC-33385P, Revision 0, November 2007. However, this section does not discuss the time necessary for the repair of systems required to achieve and maintain cold shutdown nor the increase in decay heat generation following plant trips. Please verify that the plant can meet the 72-hour requirements in both 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with increased decay heat at MUR power uprate conditions.

IV. The following question is provided from the Containment and Ventilation Branch (SCVB):

1. On page 3 of Attachment 1 to your license amendment request (LAR) dated November 19, 2007 in Section 2.0, "PROPOSED CHANGE,", there is a bulleted item:

   > ALLOWABLE VALUE on page 3.3-51 for TS Table 3.3.6.1-1, FUNCTION 1.c., Main Steam Line Flow – High, is revised from "≤ 142.7% rated steam flow."

   Given the information presented in Section 5.3.5, "Main Steam Line High Flow Isolation," on page 5-5 of the attachment NEDO-33385, Revision 0, "Safety Analysis Report for CNS TPO," to your LAR, it is not readily apparent how the proposed allowable value of "142.7%" was determined. Provide additional detail as to how the adjustment to the Main Steam Line High Flow Isolation Allowable Value was made to arrive at a value of "142.7%."

V. The following questions are provided from the Reactor Systems Branch (SRXB):

1. Please identify and justify if any of the limiting analytical values that are part of the current Analysis of Record need to be modified as a result of operation at the MUR power uprate conditions at CNS.

2. Describe the MUR power uprate core and batch size of GE fuel. If the MUR core is a mixed core, give details as required by the staff safety evaluation report for NEDC-32938P, "Generic Guidelines and Evaluations for GE BWR Thermal Power Optimization (TPO)" (Agencywide Documents Access and Management System Accession No. ML031050138). Discuss the impact of any new fuel type introduction on the proposed power uprate.

3. On page 5 of Attachment 1 of the submittal, it was stated that uncertainty in feedwater (FW) flow measurement is the most significant contributor to core power measurement uncertainty. In comparison, please discuss how significant is the boiling-water reactor recirculation flow measurement uncertainty, and justify how you assure that the uncertainty in recirculation flow measurement will not challenge the remaining uncertainty of 0.3 percent (2.0 percent - 1.7 percent) in core power measurement.

4. Please identify and discuss the significance of any differences between the actual CNS plant FW piping configuration and the model used at Alden Research Lab. The response should include deviations that can impact FW flow characteristics, such as pipe elbows, and any differences in geometry upstream of the FW flow instruments.

5. If a leading edge flow monitor (LEFM) becomes inoperative, the staff understands that the existing flow nozzles, that have been calibrated with the last valid LEFM data, will be relied upon for a short period of time. If a defouling event should occur in the existing flow nozzles during this time period, an overpower condition could result. Please discuss this possibility.

6. Please discuss the frequency of the listed preventive maintenance activities.

7. Please discuss the procedure for installation and testing of the flow measuring instrumentation, including the following additional information:

   i) An estimated time (hours or days) for installation and testing, and the potential radiation exposure to the technicians during installation and testing.

   ii) The Mode of plant operation (Modes 1 through 5) in which the plant is required (or preferred) to be for the purpose of installation and testing of the flow measuring instruments.

   iii) If the instruments (including electronics) are located in a radiation area of the plant, please discuss the impact of any expected radiation damage on the instrumentation and the resulting degradation of performance of the instruments.

VI. The following questions are provided from the Technical Specifications Branch (ITSB):

1. Explain how Surveillance Requirement (SR) 3.3.5.1, “Perform a Channel Check once per 12 hours,” is sufficient to ensure that the necessary quality of Caldon CheckPlus systems and components are maintained and that facility operation will be within safety limits. This information is needed to ensure compliance with 10 CFR 50.36(d)(3).

2. Correct the typo in Limiting Condition for Operation (LCO) 3.3.5, Required Action B.2.

3. Identify the Attachment 2 Technical Specification markup pages that should contain a requirement to reset the Actions Condition, Applicability, Required Action, SR, Applicable Modes or other Specified Conditions, or LCO limits to the original licensing basis, if the Caldon CheckPlus System is inoperable for extended periods of time. This information is needed to ensure that (1) the lowest functional capabilities established in accordance with 10 CFR 50.36(d)(2)(i) are met, and (2) the SRs established in accordance with 10 CFR 50.36(d)(3) are met for the aforementioned time periods, which are not limited in duration from the time that Mode 1 is entered.