April 19, 2004

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

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - CORRECTION TO ISSUANCE OF AMENDMENT NO. 172, STRETCH POWER UPRATE (TAC NO. MB9031)

Dear Mr. Coutu:

On February 27, 2004, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 172 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP).

Amendment No. 172 approved the KNPP 6.0-percent stretch power uprate and revised the Operating License and Technical Specifications (ADAMS Accession No. ML040430633).

Two errors were discovered in the safety evaluation (SE) after the issuance of Amendment No. 172.

One error was in Section 3.2.2.12.2.15, "Steam Generator Tube Rupture (SGTR)." This section referred to Title 10 *Code of Federal Regulations* Section 100 (10 CFR 100) limits; this is incorrect for KNPP. License Amendment No. 166, dated March 17, 2003 (ADAMS Accession No. 030210062), approved the alternate source term methodology for design-basis radiological accident analysis. Therefore, the regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50 Appendix A, General Design Criteria 19, "Control Room," as supplemented by Section 6.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP). This error did not affect the conclusion of Section 3.2.2.12.2.15, "Steam Generator Tube Rupture (SGTR)." The corrected SE pages 48 and 49 are enclosed in Enclosure 1.

The second error was in Section 3.2.2.12.1.3, "RCS Flow Rate." The following statement was made in Section 3.2.2.12.1.3: "The licensee has committed to reduce its predicted flow rate by this amount when making a calorimetric calibration of its elbow tap flow meters (Reference 31)." The licensee never made a commitment to reduce its predicted flow rate by approximately 100 gallons per minute when making a calorimetric calibration of its elbow tap flow meters. Reference 31 referred to boric acid precipitation; therefore, it is incorrect. This error did not affect the conclusion of Section 3.2.2.12.1.3, “RCS Flow Rate.” The corrected SE page 33 is enclosed in Enclosure 2.

If there are any comments or questions concerning this letter, please contact me at (301) 415-1446.

Sincerely,

/RA/

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

Docket No. 50-305

Enclosures: 1. Corrected SE pages 48 and 49
2. Corrected SE page 33

cc: See next page

gallons per minute when making a calorimetric calibration of its elbow tap flow meters. Reference 31 referred to boric acid precipitation; therefore, it is incorrect. This error did not affect the conclusion of Section 3.2.2.12.1.3, “RCS Flow Rate.” The corrected SE page 33 is enclosed in Enclosure 2.

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evaluates the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covers initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28 for ensuring that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria contained in SRP Section 15.4.8 and used to evaluate this accident include: (1) Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod, and (2) The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*.

The NRC staff previously reviewed and approved the licensee's analyses related to the RCCA ejection at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The licensee analyzed both the Framatome ANP fuel and the new Westinghouse 422V+ fuel for this event, and the results demonstrate that the new Westinghouse 422V+ fuel is more limiting. The NRC staff's review and approval of this transient for the stretch power uprate is discussed in detail in Section 2.4.2.15 of the fuel transition amendment.

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and concludes that the licensee's analyses have adequately accounted for operation of the plant for the fuel upgrade and the proposed stretch power uprate and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the proposed fuel upgrade and stretch power uprate. Therefore, the NRC staff finds the proposed fuel upgrade and stretch power uprate acceptable with respect to the rod ejection accident.

3.2.2.12.2.15 Steam Generator Tube Rupture (SGTR)

A SGTR event causes direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and SG safety or atmospheric relief valves. Reactor protection and ESFs are actuated to mitigate the accident and restrict the offsite dose within the guidelines of the 10 CFR 67 limits. The NRC staff's review covers postulated initial core and plant conditions, method of thermal and hydraulic analysis, sequence of events assuming with and without offsite power available, assumed reactions of reactor system components, functional and operational characteristics of the RPS, required operator actions consistent with the plant EOPs, and the results of the accident analysis. A single-failure of mitigating system is assumed for this event. The NRC staff review for SGTR discussed in this section is focused on the thermal and hydraulic analysis for the SGTR in order to: (1) support the review related to 10 CFR Part 67 for radiological consequences which is addressed below in this safety evaluation, and (2) confirm that there is no overfill of the SG during the mitigation of this event which could cause unacceptable radiological consequences or potential failure of the main steam system. Specific review criteria are contained in SRP Section 15.6.3.

In support of its proposed stretch power uprate, the licensee performed a SGTR thermal-hydraulic analysis for calculation of the radiological consequences. The analysis was performed using methodology consistent with the current analysis of record. The NSSS design parameters for the uprated core power of 1772 MWt are used in this analysis. Following a SGTR, a loss of off-site power is assumed to occur concurrent with the reactor trip resulting in the release of steam to the atmosphere via the SG atmospheric relief valves and/or safety valves. Consistent with the current analysis, the licensee assumes that the operators have completed the actions necessary to terminate the equilibrium break flow and the steam releases from the ruptured SG in 30 minutes after the event initiation. The resulting break flow mass transfer is then used to calculate the radiological consequences of the SGTR. In response to the NRC staff question regarding whether or not the 30 minutes operator action time for terminating the break flow is achievable, the licensee indicated that another detailed thermal-hydraulic analyses has been performed at the uprated power of 1772 MWt to evaluate the impact on the radiological consequences of the SGTR break flow continuing longer than 30 minutes and to evaluate the potential for SG overfill. This analysis used operator action times, supported by Kewaunee simulator exercises, leading to SG isolation within 30 minutes and break flow termination at approximately 49 minutes. The calculated radiological consequences are bounded by the results of the analysis assuming a constant break flow of 30 minutes. Also, the results of the analysis indicate that recovery actions can be performed to terminate the primary to secondary break flow before overfill of the ruptured SG would occur. Since the licensing analysis provided a bounding consequences of the event, the NRC staff finds the licensee’s SGTR analysis acceptable.

The NRC staff has reviewed the licensee’s analysis of the SGTR accident and concludes that the licensee’s analysis has adequately accounted for operation of the plant at the proposed stretch power uprate and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the SGTR.

3.2.2.12.2.16 Anticipated Transients Without Scram (ATWS)

ATWS is defined as an AGO followed by the failure of the reactor portion of the protection system specified in GDC-20. 10 CFR 50.62 provides the regulations regarding ATWS, and requires that:

- Each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system, and
- Each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a diverse scram system (DSS). This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

- Letdown and makeup are included as total values without consideration of which loops are actually affected.

- Pressurizer spray flow and pressurizer surge line flow are included without consideration of which loops are actually affected.

- The NRC staff's analysis was developed with an assumption that there was no mass flow into or out of the control volume, except for flow in the hot and cold legs, an assumption that may be inconsistent with the physical configuration associated with such items as letdown, makeup, RCP cooling, RCP seal injection, and the pressurizer.

The NRC staff believes that normal charging and the pressurizer spray line cross the boundary of the control volume between the T_H and T_C locations, but letdown, seal injection flow, the RCP thermal barrier cooler heat removal, and the pressurizer surge line are outside the heat balance boundary defined by the locations of T_h and T_c. For purposes of estimating $Q_{loss\Delta T}$, the NRC staff assumed the following characteristics:

- Normal charging minus seal injection = 50 gpm ≡0.8 MBTU/hr
- Pressurizer spray = 15 gpm ≡0.41 BTU/hr
- RV heat loss rate = 0.2 MBTU/hr
- Control rod drive heat loss rate = 1.5 MBTU/hr
- Hot and cold-leg pipe heat loss rate = 0.05 MBTU/hr

Normal charging introduces cool water from the regenerative heat exchanger downstream of T_c and, in effect, is a heat loss. Conversely, the pressurizer spray line does not influence T_c. Thus, $Q_{loss\Delta T}$ is approximately 0.8 + 0.2 + 1.5 + 0.05 = 2.6 MBTU/hr ≡80 gpm.

RCS flow passes through the elbow tap measurement location before the letdown stream is removed. Assuming letdown and charging rates are identical, the same flow rate is reinjected into the RCPs and cold-legs, so that total RV flow is unaffected. Conversely, the nominal 15 gpm pressurizer spray bypasses the reactor vessel but is indicated by the elbow tap instrumentation.

The NRC staff consequently estimates that the licensee's analysis over-predicts RCS flow rate by 26 + 80 + 15 ≡ 100 gpm. The licensee has said it plans to reduce its predicted flow rate by this amount when making a calorimetric calibration of its elbow tap flow meters. Since the MUR power uprate dated July 8, 2003, review was conducted with respect to a 1.4 percent power increase, a number of other considerations were identified that were not included in the NRC staff's review. They were addressed in the letter from Shukla, Girija S., "Diablo Canyon Power Plant, Unit No. 1 and Unit No. 2 - Issuance of Amendment - Revision of Technical Specification (TS) Table 3.3.1-1, 'Reactor Trip System Instrumentation,' and Revised Reactor Coolant System Flow Measurement) TAC Nos. MB6760 and MB6761," Letter from NRC to Gregory M. Rueger, Pacific Gas and Electric Company, ML032380158, August 21, 2003, review that was being conducted at the same time. These considerations include the following:

- Elbow tap flow meter coefficients remain sufficiently constant that the relative changes of flow rate through the cold-leg elbows can be correlated with the relative changes in the elbow tap ΔPs during long-term operation provided sufficient attention is paid to