September 29, 2006

Mr. Randall K. Edington
Vice President-Nuclear and CNO
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - CORRECTION TO SAFETY EVALUATION RE: AMENDMENT NO. 222 (TAC NO. MC8566)

Dear Mr. Edington:

By letter dated September 5, 2006, the Commission issued Amendment No. 222 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment replaces the current accident source term used in the fuel handling accident design-basis radiological analysis with an alternative source term pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.67, "Accident Source Term." The amendment was in response to your application dated September 29, 2005, as supplemented by letters dated January 16 and April 7, 2006.

Enclosed are the corrected safety evaluation (SE) pages 4, 6, and 7. The revisions are identified by a line in the margin.

This letter should be a supplement to our initial letter dated September 5, 2006. This change corrects the SE. We regret any inconvenience this may have caused you.

Sincerely,

/RA/

Brian Benney, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: Safety Evaluation pages

cc w/encls: See next page

September 28, 2006

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resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

The NRC staff considers the final version and the draft version of this section of NUMARC 93–01, Revision 3, to be functionally equivalent. The licensee committed to implement the guidelines of Section 11.3.6.5 of NUMARC 93–01, Revision 3, within 30 days of the implementation of this license amendment. The NRC staff finds this meets the commitment required by TSTF–51, Revision 2.

Because the licensee has applied to implement an AST selectively for the FHA, the regulatory dose criteria of 10 CFR 50.67 are used in lieu of the 10 CFR Part 100 dose limits in determining acceptability of the changes.

3.1 Radiological Consequences of the Fuel Handling Accident

The licensee evaluated the dose consequences of an FHA following both a 24-hour decay time since reactor shutdown and a 7-day decay time since reactor shutdown. This evaluation was based on the AST guidelines outlined in NUREG–1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183, as well as the NRC computer code RADTRAD Version 3.03 (NUREG/CR–6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," including Supplements 1 and 2). To support its TS changes, the licensee's evaluation demonstrated that the radiological doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the control room (CR) are within regulatory limits without crediting the operability of secondary containment, secondary containment isolation valves, the SGT system, or secondary containment isolation instrumentation after a 24-hour decay period following reactor shutdown. Likewise, the licensee's evaluation demonstrated that control room radiological doses are within regulatory limits without crediting the operability of the CREF system and CREF system instrumentation after a 7-day decay period following reactor shutdown.

The licensee's limiting postulated FHA event assumed a fuel assembly is dropped into the reactor core during refueling operations from a height of 32.95 feet, which is the maximum height allowed by the fuel handling equipment. The resulting impact of the fuel assembly onto the top of the core is assumed to damage 151, GE14, 10×10 fuel rods (representing approximately 0.3155 percent of the core) causing a gap release of radionuclides to the water pool above the core. This event could also occur over the spent fuel pool. However, the licensee states that significantly fewer (i.e., 48) fuel rods would be damaged in the latter case, due to reduced drop height. Since both the reactor cavity and spent fuel pool are located in the reactor building, an FHA in either the reactor cavity or the spent fuel pool is assumed to have the same potential release pathways from the reactor building to the environment.

Given that the primary containment remains open during refueling, the licensee assumed the radionuclides released as a result of the FHA pass through the water in the reactor cavity and enter the reactor building refueling floor atmosphere instantaneously. The water pool above the core serves as a barrier to the release of a significant amount of radionuclides. The remaining radionuclides that become airborne in the reactor building are assumed to be released to the outside environment over a 2-hour period.

The licensee's FHA evaluation of control room doses for the 24-hour decay time case credited the operability of the CREF system and CREF system instrumentation. During both normal and radiological emergency modes of operation, the control room envelope is positively pressurized and the return air from the control room envelope is recirculated without filtration. During the first 1 minute of the event, the licensee assumed a normal unfiltered inflow of 3235 cubic feet per minute (cfm). The CREF system was then assumed to actuate due to high radiation detected in the reactor building exhaust plenum.[3] For the remaining duration of the event, the licensee assumed an emergency filtered inflow of 810 cfm.[4] The licensee also assumed an unfiltered inleakage of 400 cfm throughout the entire duration of the event.[5]

The licensee's FHA evaluation of control room doses for the 7-day decay time case did not credit the availability of the CREF system. For this scenario, a normal unfiltered inflow of 3635 cfm (which includes 400 cfm inleakage) was used for the duration of the accident.

The licensee also qualitatively assessed the potential gamma shine dose from external sources to the control room during the FHA. The radiation sources external to the control room include the airborne external cloud and CREF system filters located within the control room envelope. The licensee concluded that the cloud shine and CREF system shine control room doses would be a fraction of the inhalation doses and the resulting total dose would still be below regulatory criteria.

The NRC staff reviewed the information provided in the licensee's submittal, as supplemented, and also performed an independent calculation that confirmed the licensee's dose results. The assumptions used by the licensee (which are listed in Table 1 of this SE) were found acceptable by the NRC staff. The licensee's calculated dose results are given in Table 2. The NRC staff has found the licensee's calculated doses acceptable because they are within the SRP 15.0.1 radiological dose acceptance criteria for an FHA. These criteria are 6.3 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the EAB for the worst 2 hours, 6.3 rem TEDE at the LPZ for the duration of the accident, and 5 rem TEDE in the CR for the duration of the accident.

3.2 Atmospheric Dispersion Estimates

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inoperable, and (2) justify why the reactor building vent release pathway is assumed to bound all of these other possible release pathways for the control room dose assessment. In its letter response dated April 7, 2006, the licensee identified a number of additional potential release pathways to the environment assuming an FHA on the refuel floor but stated that the reactor building vent remained bounding because of its direct path to the outside environment and relatively short distance to the control room intake. For additional conservatism, the licensee made a commitment to keep the reactor building roof hatch closed during movement of irradiated fuel in the secondary containment.

3 The licensee evaluated the ability of the reactor building exhaust plenum radiation monitor to provide the necessary CREF system initiation signal for the FHA. This setpoint assessment is predicated on having ventilation flow in the exhaust plenum at the start of the FHA. Therefore, the licensee has committed to revising the TS 3.3.7.1 basis for the CREF system instrumentation to formalize that adequate ventilation exhaust airflow is a support function to the reactor building ventilation exhaust plenum high radiation monitor during the movement of lately irradiated fuel. In this manner, ventilation exhaust airflow will be a TS operability requirement for the radiation monitor.

4 Although the CREF system is currently specified as having a 95 percent efficiency for all iodine species, the licensee modeled a filtered efficiency of 89 percent for all iodine species in its analyses in order to provide for future analytical operational margin.

5 The unfiltered inleakage assumption of 400 cfm is conservative when compared to the inleakage value of 64 cfm measured by the licensee during tracer gas testing for the radiological emergency mode of operation as reported in its in September 30, 2004, response to NRC Generic Letter 2003–01, "Control Room Habitability."

3.2.1 Control Room χ/Q Values

The licensee's control room dose analyses modeled FHA releases as occurring through the reactor building vent and assumed a single receptor point at the CR air intake for both unfiltered and filtered inflow and unfiltered inleakage. The licensee used existing CR atmospheric dispersion factors (χ/Q values) that had been submitted for NRC staff review as part of the submittal for the current FHA of record (License Amendment No.187). These χ/Q values, which are presented in Table 1, were listed in the calculation provided in Enclosure 1 to the licensee's letter dated September 14, 2001. They were generated using the NRC computer code ARCON96 (NUREG/CR–6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and onsite meteorological data collected during calendar years 1994–1998.

The release assumptions used for the currently licensed FHA and the one proposed in this license amendment request are somewhat different. The previous FHA assumed a 90-second unfiltered ground level reactor building vent release followed by an elevated filtered stack release for the balance of the 30-day duration, while the new analyses assumes a 30-day unfiltered reactor building vent ground level release. In its letter dated September 14, 2001, the licensee presented χ/Q values calculated for the reactor building vent release assuming three different values for the reactor building vent flow rate (i.e., 51,333 cfm, 9500 cfm, and 1780 cfm). The three flow rates modeled a reactor building exhaust fan coast down after a primary containment isolation system trip on high radiation in the reactor building exhaust plenum. For the new analyses, the licensee elected to use the highest (i.e., most conservative) of the three sets of χ/Q values, which are the χ/Q values associated with the lowest (i.e.,1780 cfm) reactor building vent flow rate.[(6)] Although the 1780 cfm χ/Q values are not specifically cited in the License Amendment No. 187 Safety Evaluation (as they were not applicable until after the 90-second Reactor Building ground level release had terminated), they are based on the same previously approved 1994–1998 meteorological data and ARCON96 methodology. For these reasons, the NRC staff has concluded that the CR χ/Q values presented in Table 1 are acceptable for use in the CR dose assessment submitted in support of this amendment.

3.2.2 Offsite χ/Q Values

The χ/Q values used by the licensee for the EAB and LPZ dose consequence assessments are based on ground level releases from the reactor building vent calculated using site specific inputs and methodology described in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." These χ/Q values are presented in Table 1. The calculation documenting these χ/Q values was previously submitted to the NRC staff in Enclosure 6 to a letter dated February 28, 2001. A subset of these χ/Q values (i.e., the χ/Q values listed as the 0–2 hour EAB and LPZ values) was approved by the NRC staff for use in License Amendment No. 187. The remaining EAB and LPZ χ/Q values approved in License Amendment No. 187 are not applicable to the new analyses because they represent elevated stack releases. Since the remaining offsite χ/Q values listed in Table 1 represent ground level reactor building vent releases derived using the same previously approved RG 1.3 methodology, the NRC staff has concluded that the EAB and LPZ χ/Q values presented in Table 1 are acceptable for use in the offsite dose assessment submitted in support of this amendment.

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6 Each SGT system fan has a flow rate of 1780 cfm and the reactor building exhaust fan flow rates are higher.

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