August 6, 2002

Mr. David L. Wilson
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION RELATED TO NEBRASKA PUBLIC POWER DISTRICT'S SEISMIC REEVALUATION PROPOSED TO ADDRESS COOPER NUCLEAR STATION LICENSE CONDITION 2.C.(6) (TAC NO. MB4654)

Dear Mr. Wilson:

By letter dated February 26, 2002, Nebraska Public Power District, the licensee for the Cooper Nuclear Station (CNS), submitted for the U. S. Nuclear Regulatory Commission (NRC) staff to review and approve the proposed seismic reevaluation for CNS addressing the requirements of the license condition 2.C.(6). The licensee discussed its analytical approach with the NRC staff during a telephone call conducted on May 8, 2002, and indicated that it will submit its revised approach on the CNS docket. On June 9, 2002, the licensee provided the supplemental information discussed during the May 8, 2002, telephone call. The staff has reviewed your February 26, and June 9, 2002 submittals, and has identified the enclosed request for additional information (RAI) to clarify your submittals.

The NRC staff requests your docketed response to the enclosed RAI in a timely manner to support the tight schedule of your request for review and approval.

If you have any questions regarding the enclosed RAI, please contact me promptly at (301) 415-1476.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
REGARDING COOPER NUCLEAR STATION SEISMIC REEVALUATION FOR
LICENSE CONDITION 2.C.(6)

1. In your submittal (Reference 1), you indicated that the 2.0xSSE [safe shutdown earthquake] ground response spectrum (GRS) envelopes the floor response spectra (FRS) at elevation 932'-6" in both the Control Building (CB) and Reactor Building. However, Figure 4.5 shows that the FRS at elevation 932'-6" in the CB is higher than the 2.0xSSE GRS. Explain the discrepancy. Also, provide a figure, which confirms that the 2.0xSSE GRS envelopes the FRS at elevation 903'-6" in the CB.

2. You indicated that the methodology described in NUREG/CR-6240 (Reference 2) was used to determine the seismic capacity of welded and non-welded (e.g., threaded pipe) steel piping. Indicate whether NRC has reviewed and accepted the methodology as an acceptable approach to determine the seismic capacity of the steel piping.

3. You indicated in Reference 1 that the seismic demand for outlier resolution will be 2 times the GRS in the horizontal direction and 2/3 the GRS in the vertical direction for all piping systems. The 2/3 the GRS in the vertical direction is based on an assumption that there is no amplification of the vertical seismic input ground motion by the Turbine Building (TB). Justify the TB is perfectly rigid in the vertical direction.

4. You indicated in Reference 1 that the anchor bolt capacities of Appendix C of the Seismic Qualification Utility Group-Generic Implementation Procedure (SQUG-GIP) (Reference 3) will be used for the pipe support evaluations. However, if anchor bolts exist that are not given in the SQUG-GIP, then the manufacturer's capacities will be used with a factor of safety 3.0. Discuss your justification for not using the manufacturer's recommended factor of safety.

5. You used Equation 5.9 in Reference 1 for determining the adequacy of the anchor bolt capacity. Discuss how Equation 5.9 is more conservative than the bilinear formulation given in the SQUG-GIP (Reference 3).

6. Equations 5.1a through 5.3 in Reference 1 are similar to the equations contained in the ASME *Boiler and Pressure Vessel Code*, Section III, Division 1 for Class 3 piping systems. If the ASME type equations are used for a piping evaluation, then the appropriate **i** factor (stress intensification factor) from the version of ASME Code where those equations appear should be used in the evaluation.

7. You stated that the basis for the establishment of Equation 5.3 in Reference 1 is that " ... S_A for carbon steel pipe is approximately 1.5 S which is approximately 5/8 S_y. The majority of the piping is A-106B GR. B CS with S=15000 psi and S_y=36000 psi. 2.5 S_A = (2.5 x 1.5 x 15000) = 56250 psi and, therefore, 2.5 S_A is approximately 1.6 S_y. The applied stresses are secondary; limiting the range of applied stress to less than 2 S_y insures that elastic shakedown will occur, no significant membrane stress rupture will occur, and the accumulated cyclic damage will be elastic. Therefore, given the limited number of cycles of strong motion in a Design Basis SSE (10 to 20 cycles) and that elastic cycling below the 2.0 S_y will occur, a fatigue failure due to the SAM's from one

SSE would not occur. Therefore, the 1.6 S_y secondary stress range limit used is significantly less than the upper bound limit of 2 S_y and with this limit no fatigue failures due to one SSE event would be anticipated."

However, the NRC staff has a different view on Equation 5.3. Equation 5.3 specifies the use of ½ the range of SSE anchor moments. This justification implies that the range of anchor motions is held to less than 2 S_y. Your statement is not accurate unless Equation 5.3 considers the full range of SSE. Provide your discussion with respect to the staff's view.

8. You stated in Reference 1 that "... Recent criteria and studies including Regulatory Guideline 1.61 [Damping Values for Seismic Design of Nuclear Power Plants], the ASME *Boiler and Pressure Vessel Code* Section III, Division 1, Appendix N, and NUREG/CR-0098 specify levels of damping for the SSE analysis of piping systems. In all the aforementioned documents, the basis of the determination of damping values is primarily the stress level in the component, not the basis or methodology used for response spectrum generation. That is, once a response spectrum is selected, the specified damping is based on the response of the structure under analysis in terms of fabrication methods and member stress levels. Newmark and Hall in NUREG/CR-0098, specify damping values of 2% to 3% for piping stressed to no more than ½ S_y and 5% to 7% for piping stressed to approximately the yield point. The ASME *Boiler and Pressure Vessel Code*, Section 111, Division 1, Appendix N, currently specifies 5% damping for the evaluation of the piping systems at both the Level B and Level D conditions. The Level D condition corresponds to the SSE event under evaluation here."

   The NRC staff does not agree with your statement. The basis for staff acceptance of 5 percent damping is the conservatism in the spectra generation. This position has been previously stated in the NRC endorsement of Code Case N-411 in Regulatory Guide 1.84 [Design and Fabrication Code Case Acceptability-ASME Section III Division I].

9. You indicated in Reference 1 that an approach called the "collapsed beam" approach is used for localized evaluation of piping systems. The NRC staff is not aware of the "collapsed beam" approach and did not endorse the approach previously. Justify the reasons why the "collapsed beam" approach is equivalent to or more conservative than the analysis methods discussed in Sections 3.9.1 and 3.9.2 of the NRC Standard Review Plan.

10. During the teleconference held on May 8, 2002, the licensee indicated that the piping support components at Cooper Nuclear Station (CNS) are designed in accordance with the requirements in MSS-SP-58, "Pipe Hangers and Supports - Materials, Design, and Manufacture." In Reference 1, the licensee indicated that the capacities of the piping support components for the Level D load case should not exceed 2.0 times the capacities specified in MSS-SP-58 based on the ASME *Boiler and Pressure Vessel Code* Case N-500-1. The NRC staff requests response to the following:

    (a) The ASME *Boiler and Pressure Vessel Code* Case N-500-1 specified other requirements (e.g., materials, quality assurance program, etc.) in order to use 2.0 times the capacities specified in MSS-SP-58 for the Level D load case. Indicate whether the piping support components at CNS meet the pertinent

requirements of the ASME Code that would permit an increase in the load capacity by a factor of 2.0 times at the load Level D.

(b) In Reference 4, the licensee indicated that CNS Updated Safety Analysis Report specifies the use of 0.9 S_y as the stress limit for the piping support components for the Level D load case. This limit exceeds 2.0 times the capacities specified in MSS-SP-58. Provide justification for suggesting to use an even higher limit than those permitted in the ASME *Boiler and Pressure Vessel Code* Case N-500-1. Also, indicate whether NRC had reviewed and accepted your use of 0.9 S_y as a stress limit for the piping support components at CNS for the Level D load case.

11. In Reference 4, the licensee indicated that a numerical technique (i.e., finite element analysis) will be used to establish the capacities of the pipe support components. Discuss your rationale for concluding that a finite element analysis, which relies on approximation of the geometry, can be considered to provide a more realistic estimate of the load carrying capacity of the analyzed component than the actual testing performed by the vendor for such component.

12. In Reference 4, the licensee indicated that it will use the concrete anchor bolt capacities used in IE Bulletin 79-02 [Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts] with a factor of safety of 4. However, IE Bulletin 79-02 requires a factor of safety larger than 4 for certain types of anchor bolts. Provide your technical justification for using only the factor of safety of 4.

References:

1. Letter, Nebraska Public Power District to U.S. NRC, “License Condition 2.C.(6) Seismic Evaluation, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46,” dated February 26, 2002.

2. NUREG/CR-6240, Stevenson & Associates, “Application of Bounding Spectra to Seismic Design of Piping Based on the Performance of Above Ground Piping in Power Plants Subject to Strong Motion Earthquakes,” February, 1995.

3. Seismic Qualification Utility Group, “Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment,” Revision 2, corrected February 14, 1992.

4. Letter, Nebraska Public Power District to U.S. NRC, “Supplemental Information Related to License Condition 2.C.(6) Seismic Evaluation, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46,” dated June 9, 2002.

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