May 20, 2004

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

SUBJECT: COOPER NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK-INFORMED RELIEF REQUEST RI-34 (TAC NO. MC2351)

Dear Mr. Edington:

By letter dated March 11, 2004, Nebraska Public Power District (the licensee) requested the Nuclear Regulatory Commission (NRC) staff grant relief from certain inservice inspection (ISI) requirements for the selection and examination of Class 1 and 2 piping welds. The submittal proposes a Risk-Informed ISI Program as an alternative to existing American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements.

The NRC staff has reviewed the information provided in the March 11, 2004, submittal and determined that additional information is required in order to complete the review of RI-34. As agreed upon with Bill Victor of your staff on May 18, 2004, the licensee will respond to the request for additional information (RAI) within 45 days. The RAI is enclosed.

Sincerely,

/RA/

Michelle C. Honcharik, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosure: RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

ISSUES RELATED TO RISK-INFORMED RELIEF REQUEST RI-34

COOPER NUCLEAR STATION (CNS)

Unless otherwise stated, all Table, Section, and Page numbers refer to Enclosure 1 of the March 11, 2004, submittal.

1. Regulatory Guide (RG) 1.178, *An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping,* Revision 1, dated September 2003, replaced the original "*For Trial Use*" RG dated September 1998. Revision 1 of the RG 1.178 includes guidance on what should be included in risk-informed inservice inspection (RI-ISI) submittals, particularly in dealing with probabilistic risk assessment (PRA) issues. Specifically, on Page 28 of RG 1.178, the following is stated regarding the information that should be included in a submittal:

   > A description of the staff and industry reviews performed on the PRA. Limitations, weakness, or improvements identified by the reviewers that could change the results of the PRA should be discussed. The resolution of the reviewer comments, or an explanation of the insensitivity of the analysis used to support the submittal to the comment, should be provided.

   Section 1.2 discusses the CNS Individual Plant Evaluation (IPE), noting that the NRC approved its results in a letter dated May 18, 1995, concluding that the IPE had met the intent of Generic Letter 88-20 and had identified plant-specific vulnerabilities per the guidance of NUREG-1335.

   a. Our review of the NRC Staff Evaluation for your IPE appears to indicate no weaknesses with that document. Please confirm that this is your understanding or indicate 1) what weaknesses were identified and 2) what was done to correct the identified weaknesses, or why the uncorrected weaknesses are not relevant to this application.

2. Section 1.2 notes that as a result of the two industry reviews of the CNS PRA, you are currently performing a major revision to the PRA and that this particular revision was not used in the preparation of your submittal. From this discussion, four areas of your PRA were reported to either be in revision or are "opportunities for improvement":

   - Internal Flooding Analysis (from the first of the two peer reviews)
   - Initiating Event (IE) Analysis (from the latter of the two peer reviews)
   - Data Analysis (from the latter of the two peer reviews)
   - Human Reliability Analysis (HRA) (from the latter of the two peer reviews)

   For the first item you noted that "certain conclusions regarding internal flooding were considered qualitatively and reviewed against the most current plant information for potential insights." You contended that the latter three items should not impact the consequence rankings determination, mostly because the "the risk importance of the systems in the RI-ISI process is dominated by the LOCA events.” As discussed below, the NRC staff requests additional information on the first (flooding) and the fourth (HRA) items.

The NRC staff concurs with your judgment that revisions to the IE Analysis should have very little influence on the consequence rankings of pipe segments, and hence, on this PRA application, because the Conditional Core Damage Probability (CCDP) for pipe breaks that cause IEs is determined by dividing out the initiating event frequency (IEF) term, leaving CCDP unchanged, regardless of IEF changes and for pipe breaks that impact mitigating capability, the CCDP should not be drastically affected by the “composite” change in IEF terms, since the change in CCDP will be proportional to the difference in IEFs in each one of the “delta cutsets”. Barring a highly unusual trend, changes in IEF are expected to be either small or negative (current trend of events in the industry continues to decline).

The NRC staff also concurs with your judgment that revisions to the Data Analysis should have very little influence on the consequence rankings of pipe segments, primarily because it is likely that a revised Data Analysis will generally reflect both increases and decreases in the availability and reliability of equipment, resulting in offsetting impacts on core damage frequency (CDF) calculations (both for the case where the pipe segment’s surrogate Basic Event is set to TRUE as well as for the base case). However, even if there were a significant change in the performance of certain equipment in recent years, the impact on the Data Analysis (i.e., the Basic Event values) would be dampened by the Bayesian Update process. Because of overall improvements in equipment performance in recent years, there is likely a bias toward lower CDFs, and hence, slightly lower CCDPs. Thus, it appears that there is little chance that revisions to the Data Analysis would impact the CCDP of any pipe segment to the extent that its consequence ranking would need to be elevated.

a. The first area of improvement listed above suggests that you are preparing to incorporate an upgraded internal flooding model into your PRA. Please explain the above statement about “certain conclusions regarding internal flooding were considered qualitatively....” (from Page 3 of 29). Please provide a description of the conclusions you have considered qualitatively and explain why they have no impact on this RI-ISI application, including a discussion about any new flooding propagation or spatial impact scenarios that may also be applicable to the consequences of pipe breaks being considered in this application.

b. Significant changes to the HRA could be influential. Revised HRA methodology can sometimes cause significant revisions to Human Error Probabilities (HEPs), which are not dampened by a Bayesian process. When several of the dominant cutsets of your PRA contain the same Human Error Basic Event and equipment Basic Event used as a surrogate for a specific pipe segment and a revised HRA significantly increases this HEP, it could result in a significant increase to the CCDP of that pipe segment, resulting in an elevated consequence ranking. Please provide additional information to support your contention that these changes should not impact the consequence rankings.

3. The paragraph at the bottom of Page 2 of 29 refers to a 1996b Level 2 PRA model. Please confirm that it was intended to refer to a Level 1 PRA model.

4. Section 3.6.1 indicates that you used the “Simplified Risk Quantification Method” as described in Section 3.7 of the Electric Power Research Institute (EPRI) topical report (TR)-112657, in support of your overall risk impact assessment. You selected a value of 1E-08 per weld-year as the pressure boundary failure (PBF) frequency for a weld with no known degradation mechanism (i.e., low failure potential) and a value of 20 times that (i.e., 2E-07) for a weld with medium failure potential, which is similar to the failure rate used by a couple of the pilot plants for the EPRI TR-112657, as noted by your citation of References 9 and 14 in EPRI TR-112657.

    a. Given this information, as an example, a Category 4 weld should have a contribution to CDF of (1E-3)*(1E-8) or 1E-11/year. Assuming that the inspections are 100 percent effective in finding flaws before they progress to a rupture, then the decrease of one weld inspection should result in an increase in CDF of 1E-11/year. Table 3.6-1 (as well as the Table on Page 12 of 29) which present the risk impact results, indicates a net decrease of one system NB category 4 weld inspection, resulting in a CDF increase of 5E-12/year. Please clarify this apparent discrepancy.

    b. Many of the numerical entries in Table 3.6-1 have the same CDF or large early release frequency (LERF) values in the “w/ POD [probability of detection]” column as in the “w/o POD” column (most of these are Category 4 welds). Some of the entries have different CDF and LERF values. Please explain why sometimes the “w/ POD” and “w/o POD” values are different and sometimes they are the same.

cc:

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