February 21, 2003

Mr. Clay C. Warren
Vice President-Nuclear/Chief Nuclear Officer
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT REGARDING DESIGN BASIS ACCIDENTS' RADIOLOGICAL DOSE ASSESSMENT METHODOLOGIES, AND REVISION TO LICENSE CONDITION 2.C.(6) (TAC NO. MB4654)

Dear Mr. Warren:

The Commission has issued the enclosed Amendment No. 196 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of approval of changes to CNS design-basis accident evaluation methodology, and revision to License Condition 2.C.(6) in response to your letter dated February 28, 2001, as supplemented by letters dated February 26, September 13 and 27, and November 25, 2002 (2). Your February 26, 2002, letter was the continuation of your request dated February 28, 2001, which was partially completed by Amendment No. 187, dated October 23, 2001.

The information provided in your letter dated February 26, 2002, and the supplements provided additional information of clarifying nature, and did not expand the scope of the application originally noticed in the *Federal Register*. It did not change the U. S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in *Federal Register (66 FR 48289,* dated September 19, 2001*).*

In Amendment 187 to License No. DPR-46, the NRC staff approved the revised fuel handling accident methodology, and provided interim approval of revised dose assessment methodologies for the loss-of-coolant accident (LOCA), and control rod drop accident (CRDA). The approval was contingent upon the licensee maintaining the ability to monitor radiological conditions during emergencies and administering potassium iodide to the control room operators to maintain radiological exposure doses below the guidelines of General Design Criterion (GDC)-19, "Control Room." The review of the main steamline break (MSLB) accident dose assessment methodology was deferred to allow the licensee to complete seismic analyses of the main steamline isolation valve (MSIV) leakage pathway. The licensee provided the requested MSIV leakage pathway seismic analyses by letter dated February 26, 2002, and its supplements discussed above.

This amendment approves: (1) the dose assessment methodology for MSLB accident and CRDA, (2) the remaining meteorological assessments for the approval of LOCA and the CRDA dose assessment methodologies, and (3) on an interim basis, the LOCA dose assessment methodology, for one additional fuel cycle. The final approval of the LOCA radiological dose assessment methodology is deferred pending approval of design and completion of modification of the MSIV leakage pathway.

The amendment also approves the proposed methodology for evaluating the seismic adequacy of the piping from the MSIVs to the main turbine condenser, the turbine condenser, and the turbine building (TB). The NRC staff did not review, in detail, the specifics relating to the proposed modifications to MSIV leakage pathway systems, because they were revised again by a letter dated December 19, 2002. The review of the licensee's submittal of December 19, 2002, for a revised proposed modification of MSIV leakage pathway configuration is being conducted as a separate action. In the interim, the NRC staff has approved revised License Condition 2.C.(6) proposed in the licensee's letter of November 25, 2002. The revised License Condition 2.C.(6) replaces the current License Condition 2.C.(6).

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

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

Enclosures: 1. Amendment No. 196 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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OFFICE	PDIV-1/PM	PDIV-1/LA	EMEB/BC*	SPSB/BC*	OGC	PDIV-1/SC
NAME	MThadani:sab	MMcAllister	GImbro	MJohnson	TSherwin	RGramm
DATE	2/19/03	2/13/03	01/23/2003	01/23/2003	2/20/03	2/20/03

OFFICIAL AGENCY RECORD

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee), dated February 28, 2001, as supplemented by letters dated February 26, September 13 and 27, and November 25, 2002 (2), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly:

   A. The license is amended to authorize revisions to the Safety Analysis Report to reflect the changes to the calculation methodology for assessing the radiological consequences of design-basis accidents as approved in the enclosed safety evaluation. This authorization will expire upon CNS entering Mode 4 of the refueling outage 22.

   B. The license is amended by changes to Paragraph 2.C.(6) of the Facility Operating License No. DPR-46, as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 21, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 196

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following page of the Facility Operating License DPR-46, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
4	4

2.C.(4) Fire Protection

The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1 through 3.37 of the NRC's Fire Protection Safety Evaluation (SE), dated May 23, 1979, for the facility. These modifications will be completed prior to July 1, 1980.

In addition, the licensee shall submit the additional information in Table 3.1 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

The licensee is required to implement the administrative controls identified in Section 6 of the SE. The administrative controls shall be in effect by November 1, 1979.

C(5) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 178, are hereby incorporated into this license. Nebraska Public Power District shall operate the facility in accordance with the Additional Conditions.

C(6) Upon receiving NRC approval of the licensee's seismic evaluation of the main steam isolation valve leakage pathway to the main turbine condenser, the main turbine condenser, and the turbine building, the licensee shall fully implement the approved request, including the associated modifications, prior to restart from refueling outage 22. Until implementation is completed, potassium iodide will continue to be made available to Control Room personnel during a loss-of-coolant accident with core damage.

D (Not used)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 196

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated February 28, 2001, as supplemented by letters dated February 26, September 13 and 27, and November 25, 2002 (2), Nebraska Public Power District (the licensee) requested approval of its proposed revisions to its design-basis accident (DBA) assessment methodology, and proposed changes to its Operating License Condition 2.C.(6). On October 23, 2001, the Commission issued Amendment No. 187, as a partial response to the application of February 28, 2001. The February 26, 2002, letter was submitted in support of a continuation of the remainder of the reviews of the February 28, 2001, application that were deferred in Amendment No. 187.

In Amendment 187 to DPR-46, the U. S. Nuclear Regulatory Commission (NRC) staff approved the revised DBA assessment methodologies for the fuel handling accident (FHA), and provided interim approval of the revised DBA methodologies for the loss-of-coolant accident (LOCA) and control rod drop accident (CRDA) for one operating cycle (Cycle 21). The NRC staff identified issues that required additional analyses related to the proposed methodology for assessment of the LOCA, CRDA, and main steam line break (MSLB) accidents. The NRC staff retained the existing License Condition 2.C.(6), which required the licensee to submit a seismic evaluation of the main steamline isolation valve (MSIV) leakage pathway, and maintain the ability to monitor radiological conditions during emergencies and administer potassium iodide to the control room operators to maintain doses within the guidelines of General Design Criterion (GDC)-19, “Control Room.”

The licensee's letter dated February 26, 2002, and its supplements discussed above, submitted the additional meteorological analyses and seismic analyses of MSIV leakage pathway piping from the MSIV to the main turbine condenser, the main turbine condenser, and the turbine building (TB).

The NRC staff's review of the licensee's revised submittals for MSLB accident radiological consequences analysis, the resolution of meteorology open items for the analysis of the LOCA and CRDA, and changes to Operating License Condition 2.C.(6) are evaluated below.

2.0 REGULATORY EVALUATION

2.1 Dose Assessment Methodology

The regulatory requirements on which the NRC staff based its acceptance are Title 10 of the *Code of Federal Regulations* (10 CFR) Section 100.11 and 10 CFR Part 50, Appendix A, GDC-19. The NRC staff referred to guidance in its review of the DBA MSLB radiological consequences analysis from NUREG-0800 "Standard Review Plan" (SRP) Section 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)." This section denotes the regulatory acceptance criterion for the MSLB radiological consequences analysis to be a small fraction (i.e., 10 percent) of the Part 100 guideline values for an assumed Technical Specifications (TSs) equilibrium iodine concentration in the reactor coolant, while the Part 100 guideline values (i.e., 100 percent) are the acceptance criteria for an MSLB with an assumed iodine concentration in the reactor coolant equal to the TS maximum concentration. The NRC staff also referred to SRP Section 15.4.9, Appendix A, "Radiological Consequences of Control Rod Drop Accident (BWR)." This section denotes the regulatory acceptance criteria for the CRDA radiological consequences analysis that are well within (i.e., 25 percent of) the Part 100 guideline values. GDC-19 is the applicable requirement for the control room with regard to the radiological consequences of DBAs.

2.2 MSIV Leakage Pathway Seismic Ruggedness

The credit for dose consequence mitigation by the MSIV leakage pathway is contingent upon reliable direction of the MSIV leakage to the main turbine condenser. The NRC staff bases its acceptance on the confidence that the MSIV leakage pathway and its associated components and equipment must remain intact to direct the leakage to the main turbine condenser during DBA conditions concurrent with a postulated safe shutdown earthquake (SSE).

3.0 TECHNICAL EVALUATION

3.1 Evaluation of the Proposed DBA Dose Assessment Methodology

The licensee evaluated the impact of the proposed changes in methodology to show that applicable regulatory acceptance criteria would continue to be satisfied. The discussion below identifies the inputs and assumptions provided by the licensee and utilized by the NRC staff to perform independent calculations. The results of the NRC staff's independent calculations and evaluations were used to determine the acceptability of the licensee's analytical methodologies.

The design inputs utilized by the NRC staff to evaluate these accidents are listed in Tables 1 and 2. Inputs and model development issues that warrant further discussion are given below.

3.1.1 Main Steam Line Break Radiological Consequences Analysis

The licensee performed a re-analysis of the radiological consequences of the MSLB considering two cases for reactor coolant activity concentration. Doses were calculated for a MSLB with the reactor coolant at both the TS reactor coolant equilibrium activity concentration of 0.2 µCi/gm Dose Equivalent Iodine-131 (DEI-131) and at the TS reactor coolant maximum activity concentration of 4.0 µCi/gm DEI-131. This analysis was based on the methodology of the existing Cooper Nuclear Station (CNS) Updated Safety Analysis Report Section 14.6.5 dose analysis, with changes as discussed below.

In place of the licensing basis plume release offsite relative concentration (X/Q) values, the licensee used puff release X/Qs, based on the methodology given in Regulatory Guide (RG) 1.78, Rev. 0, Appendix B, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." Its assumptions are identical to assumptions made in a common derivation of the steady-state, straight-line Gaussian plume model. The NRC staff made comparative calculations by integrating the Gaussian puff model for a ground-level release during puff passage over a point to estimate an effective exposure, and obtained resultant dose estimates comparable to those calculated by the licensee.

The licensee calculated the dose to the control room operators as a result of a MSLB with the reactor coolant at the TS equilibrium activity concentration limit of 0.2 µCi/gm DEI-131. Rather than modeling the expected operation of the control room ventilation systems, the licensee conservatively modeled the accident release activity by taking into account that the control room is trapped, the control room isolates after 60 seconds, and the control room ventilation system remains on a lower intake rate (and clearing rates) for the duration of the accident. No filtration of the intake air is assumed. Additionally, the licensee conservatively assumed that the puff stayed at ground level with the maximum hemispherical diameter passing the control room air intake. In place of the licensing basis control room X/Q values, the licensee entered the inverse of the intake rate into its dose calculation computer code. The NRC staff finds that the licensee's conservative assumptions and calculation inputs are acceptable.

The licensee's dose results are given in Table 4, which also shows that the calculated doses meet the SRP dose acceptance criteria for offsite doses, and also meet GDC-19 dose criteria in the control room. Using the information provided by the licensee and using guidance in SRP Section 15.6.4, the NRC staff performed independent calculations for both cases of the MSLB and confirmed the licensee's results.

In a request for additional information (RAI), dated May 28, 2002, the NRC staff asked why the licensee did not perform control room dose analysis of the MSLB at the TS maximum activity concentration of 4.0 µCi/gm DEI-131. In a response dated September 19, 2002, the licensee stated that the current CNS control room licensing basis does not require analysis of the MSLB at the TS specific activity maximum allowable value. Because of this and because the licensee has not requested changes to control room systems or the MSLB source term, the NRC staff does not find that an analysis of control room doses from an MSLB with the reactor coolant at the TS maximum activity concentration is necessary for this amendment request.

3.1.2 Meteorology Considerations

The licensee used onsite meteorological data collected during calendar years 1994 through 1998 for those control room X/Q values calculated using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The data were previously discussed as part of the safety evaluations (SEs) related to Amendments 183 and 187, in which the staff commented upon several aspects of the data. As part of the instant amendment request, the licensee provided additional information about the data, as well as a description of a series of measures initiated in 1998 to improve the measurements program. The NRC staff has determined that the data is adequate for use in the dose assessments described herein, but recommends that data from the improved program be considered for use in any future calculations.

For a postulated LOCA or CRDA release from the TB, the licensee used ARCON96 to calculate control room X/Q values assuming a diffuse source. The licensee performed X/Q calculations for both the loss-of-offsite power (LOOP) and non-LOOP case for the LOCA and determined that the LOOP case assuming a diffuse release from the TB was more limiting than assuming a non-LOOP point release from the design ventilation plenum. The NRC staff has published draft RG DG-1111, “Atmospheric Relative Concentrations for Control Room Habitability Assessments at Nuclear Power Plants” in the time since the licensee originally performed the calculations. Although the licensee and its contractor did not have the draft staff guidance available to use in calculating the control room X/Q values for a diffuse source, the calculation deviated from the draft guidance in only a few respects. The primary difference was the assumption of the TB as a diffuse source; other differences included the formulation of the cross-sectional area and the distance assumed for the diffuse source, and the averaging sector width constant.

The NRC staff has evaluated the licensee’s use of the assumption of a TB diffuse source in the calculation of the control room X/Q values. The licensee provided detailed information on the location, size, and characteristics of possible openings in the TB wall to the environment, which the NRC staff used in its assessment. DG-1111 gives the NRC staff’s position that diffuse source modeling should only be used for those situations in which the activity being released is homogeneously distributed throughout the building and the assumed release rate from the building surface would be reasonably constant over the surface of the building. Although it is unlikely that the LOCA MSIV leakage and CRDA condenser leakage releases are homogeneously distributed throughout the TB, there is also no reason to assume preferential dispersion to any location on the TB wall or roof. The assumed MSIV and condenser leakage for the LOCA and condenser leakage for the CRDA are small amounts (i.e., 10 scfh for the MSIV leakage and 1 percent of condenser volume per day for the CRDA) into a large free volume in the TB. Although no credit is taken for mixing or plateout in the TB volume, the path from the leakage site within the TB to the building release point to the environment provides for mixing and diffusion, especially with a low leakage rate. There are no driving forces to cause the radioactivity to be forced out to a particular point in the TB wall or roof, and the licensee states that the CNS TB does not have roof vents to allow release of radioactivity from the TB through thermal rise. Additionally, there are no large openings directly to the environment (e.g., louvered walls) resulting in more limiting X/Q values than those calculated by the licensee using its assumptions for a diffuse release. For the reasons described in the preceding discussion, the NRC staff agrees that there is no preferential point where the release is likely to occur from the TB, and the staff has determined that the licensee’s assumption of a diffuse source for release from the TB for low level releases is acceptable. Conversely, the use of the diffuse source assumption in determining control room X/Q values for the MSLB accident would not be acceptable due to the high energy characteristics of the MSLB release.

The NRC staff has evaluated the licensee’s calculation of the control room X/Q values assuming a diffuse source for a release from the TB. The NRC staff draft guidance in DG-1111 (which was not yet written when the licensee’s calculations were performed) directs the user to determine the area of the diffuse source by finding the maximum vertical and horizontal dimensions of the above-grade building cross-sectional area perpendicular to the line of sight between the geometric center of the source building and receptor point (control room intake).

The distance is measured from the receptor point to the closest point on the building along the line of sight between the receptor and the geometric center of the source building. The licensee did not follow this formulation, but instead used the TB above-grade wall area as the source area and the shortest distance to the TB wall from the control room intake. The NRC staff has determined that the licensee's formulation of the source area and distance from the receptor and averaging sector width constant is acceptable because the licensee's assumptions, taken in total, result in X/Q values similar to those calculated using the procedures described in the draft guidance.

Therefore, on a case-specific basis, the NRC staff has determined that the licensee's use of the diffuse source X/Q values is acceptable at CNS in the current amendment request for modeling the dispersion of low level (i.e., 10 scfh) MSIV leakage during a LOCA and also for modeling the dispersion of the 1 percent volume per day leakage from the condenser for the CRDA.

3.1.3 CRDA

In CNS License Amendment Nos. 183 and 187, the NRC gave interim approval of the design basis CRDA radiological consequences analysis. The NRC staff identified unresolved issues with the licensee's use of a diffuse source methodology determining the control room X/Q values for a release from the TB, and use of AEC Technical Information Document TID-14844 to determine the core inventory for high burnup fuel. In the analysis submitted by letter dated February 28, 2001, the licensee replaced the TID-14844 based core inventory source term with the core inventory source term generated by General Electric (GE) for GE14 fuel using the isotope generation and depletion code ORIGEN2, incorporating the boiling water reactor (BWR) extended burnup library BWRUE. This change resolved an outstanding issue regarding the use of the TID-14844 source term to model extended burnup fuel as described in the NRC staff's SE accompanying Amendment 183. CNS also increased the radial peaking factor for GE14 fuel used in the CRDA analysis to its maximum expected value of 2.0 to provide adequate margin in support of future fuel operating cycles. All other inputs and assumptions are unchanged from the licensee's analysis as given interim approval in Amendments 183 and 187.

The NRC staff has reviewed the licensee's analysis of the CRDA radiological consequences, provided by letter dated February 28, 2001. The NRC staff has determined that the open item on the use of the diffuse release X/Q values to calculate the dose in the control room from a release from the TB is acceptable, as discussed above in Section 3.1.2. The NRC staff generally finds the use of isotope generation and depletion codes to be acceptable for development of the core inventory source term and has determined that the licensee's use of the core inventory source term generated by GE is acceptable. The licensee's use of a bounding maximum radial peaking factor in the calculation of the CRDA dose consequences is also found acceptable by the NRC staff since it bounds future fuel operating cycles that use GE14 fuel. The NRC staff has determined that the licensee's analysis of the radiological consequences of the CRDA generally follows guidance given in SRP Section 15.4.9, Appendix A, and finds the licensee's dose analysis of the CRDA to be acceptable. The licensee's results are given in Table 4, which also shows that the calculated doses meet the SRP dose acceptance criteria of 25 percent of Part 100 dose guidelines for offsite doses, and also meet the GDC-19 dose guidelines for the control room.

3.1.4 LOCA and Compensatory Measures

In CNS Amendments 183 and 187, the NRC gave interim approval of the design basis LOCA radiological consequences analysis, pending NRC approval of the seismic evaluation methodology used by the licensee to support taking credit of a seismically rugged MSIV leakage pathway to the main turbine condenser. Another issue unresolved in these previous amendments was the licensee's use of a diffuse source methodology in determining the atmospheric dispersion factors for a release from the TB to be used in the calculation of dose in the control room due to a LOCA. This second issue has been resolved as discussed above in Section 3.1.2 of this SE.

The licensee has committed to administer potassium iodide (KI) to the control room operators during a potential LOCA involving an indication of core damage. Based on that commitment, the NRC staff finds adequate basis for interim approval of the licensee's methodology for evaluating the LOCA dose consequences for CNS for one additional fuel cycle (until CNS enters Mode 4 following refueling outage 22). The final approval of the LOCA radiological consequences assessment methodology is deferred until the NRC staff and the licensee resolve pending issues concerning the seismic adequacy of the main steam line piping, main turbine condenser, and the TB. This issue is discussed further in Section 3.2 under "MSIV Leakage Pathway Ruggedness."

3.1.5 Control Room Habitability Generic Issue

The NRC staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room infiltration rates assumed by licensees in analyses of control room habitability. About 30 percent of power plant control rooms have been tested using enhanced test methods. In all but one case, the measured infiltration rates exceeded the values assumed in the design basis analyses. While in each case the affected licensee was able to either reduce the excessive infiltration or show the acceptability of the observed infiltration, the collective experience has caused concerns regarding facilities that have not performed the enhanced testing. The NRC staff is currently working to resolve these concerns.

Based on a review of the material discussed above, the NRC staff has determined that there is reasonable assurance that the CNS control room will be habitable during DBAs and that this amendment may be approved before the resolution of this generic issue. The approval of this amendment does not exempt CNS from regulatory actions that may be taken in the future, as this generic issue is resolved.

3.1.6 Conclusions Regarding Dose Assessment Methodology

Based on the considerations discussed above, the information provided by the licensee regarding the LOCA, the CRDA, and the MSLB, and the licensee's continuing commitment to provide KI to control room personnel, the NRC staff finds reasonable assurance that the postulated radiological consequences of the design basis LOCA, CRDA, and MSLB will be less than the dose guidelines of 10 CFR Part 50, Appendix A, GDC-19, and Section 6.4 of NUREG-0800. The NRC staff also finds reasonable assurance, based upon the considerations discussed above, that the postulated radiological consequences of the design basis LOCA, CRDA, and MSLB are within the guidelines of 10 CFR Part 100, and Sections 15.4.9 and 15.6.4

of NUREG-0800. Therefore, the NRC staff finds that the proposed changes to the CNS License Condition 2.C.(6) are acceptable with regard to the radiological consequences of DBAs.

3.2 MSIV Leakage Pathway Ruggedness

The licensee submitted information in conjunction with the implementation of License Condition 2.C.(6) at CNS on February 26, 2002 (Reference 1) and June 9, 2002 (Reference 2). In its correspondence to the NRC, the licensee identified modifications that are being considered for the proposed MSIV leakage pathway system, and discussed the evaluations required to complete the design for these modifications, which the licensee intends to finalize following receipt of the NRC's approval of the proposed evaluation methodology. The NRC staff reviewed the information provided by the licensee, and issued RAIs on August 6 and October 21, 2002. The licensee responded to the RAIs on September 27 and November 25, 2002, respectively (References 3 and 4). This evaluation addresses the seismic adequacy of the main steam line piping from the MSIVs to the main turbine condenser and the TB at CNS for the proposed MSIV leakage pathway system.

3.2.1 Seismic Demand

3.2.1.1 SSE

The SSE for CNS is based on the N69W component of the 1952 Kern County earthquake recorded at Taft, California, scaled to a peak ground acceleration (PGA) of 0.2g. The vertical PGA is taken as 2/3 of the horizontal PGA as stated in the CNS Updated Safety Analysis Report (Reference 5).

3.2.1.2 Location of Equipment and Piping

The licensee stated in Reference 1 that the majority of the piping and equipment in the scope of the MSIV leakage pathway evaluation at CNS is in the TB. The balance of the equipment is in the Reactor Building (RB) steam tunnel and torus compartment.

3.2.1.3 TB

The licensee states in References 2 and 3 that its contractor, Stevenson & Associates, generated a set of new floor response spectra (FRS) for the TB. The new FRS were generated using the soil-structure interaction analysis method and RG 1.60 ground response spectrum (GRS) anchored to the CNS SSE of 0.2g. The FRS were developed in accordance with the guidance in NUREG-0800 (NRC SRP), Sections 3.7.1 and 3.7.2. The licensee states in Reference 4 that it is in the process of reviewing the FRS developed by the contractor. During a teleconference call on November 6, 2002, the licensee indicated that the FRS were developed for inclusion in the plant licensing basis under the process described in 10 CFR Section 50.59.

3.2.1.4 RB

The licensee stated in Reference 2 that 2.0 X SSE GRS is used as the seismic demand for all elevations in the RB up to and including 932'-6". This is based on the NRC's acceptance of the

Generic Implementation Procedure for USI A-46 (GIP-2), Section 4.2.4, which states that realistic, medium centered FRS based on the IPEEE Review Level Earthquake (RLE) can be scaled to obtain realistic, median centered FRS derived from the site SEE. This is acceptable to the NRC staff.

3.2.1.5 Piping and Pipe Support

The licensee stated that the criteria established in NEDC-31858P, submitted by the BWR Owners Group (BWROG)(Reference 6) and approved in the NRC SE (Reference 7), are used for evaluation of the piping welds. For piping with non-welded joints (e.g., threaded), a one-third (1/3) reduction in the allowable unsupported spans is applied to the piping systems containing threaded fittings. The NRC staff has accepted the 1/3 reduction in the allowable unsupported spans for piping systems containing threaded fittings for other MSIV leakage pathway applications. Piping systems containing friction or cast iron components are classified as outliers and require a more detailed review and evaluation. The NRC staff found the licensee's described criteria for seismic demands for various piping runs in the MSIV leakage pathway to be acceptable.

3.2.1.6 Equipment

The licensee stated that equipment was evaluated according to the requirements of the generic implementation procedure (GIP)-2, which was accepted by the NRC staff in the resolution of USI A-46. The CNS was included in the population of plants affected by USI A-46, and has completed the actions specified in GL 87-02 for implementation of USI A-46.

3.2.1.7 Condenser

The CNS condenser is located below grade at the lowest level of the TB (Elevation 877.5 ft.). The applied seismic demand is the CNS SSE ground response spectra (GRS), which is conservative for use in the structures situated below grade.

3.2.2 Seismic Capacity

3.2.2.1 Seismic Verification Walkdowns

In order to confirm the functional capability of the leakage pathway, the licensee performed a seismic verification walkdown of the MSIV leakage pathway. The licensee indicated in its submittal (Reference 1) that the walkdown was performed by "seismic capability engineers" as defined by the "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Rev. 3." Data gathered during these walkdowns was used as input for seismic evaluations and analyses, and for identifying potential seismic interaction concerns.

The NRC staff found the walkdowns performed by the licensee, as well as the corrective actions taken for the identified outliers, to be acceptable.

3.2.2.2 TB

The seismic integrity of the TB that houses the leakage pathway piping systems and equipment was evaluated. In Reference 1, the licensee states that the majority of the piping and equipment in the scope of the MSIV leakage pathway evaluation at CNS is in the TB. The TB is built with reinforced concrete from the ground up to the operating floor and the remaining portion of the TB above the operating floor is constructed with structural steel. The licensee performed an evaluation of the TB concrete structure to determine its capability of remaining structurally intact without gross structural failure following a postulated SSE. Based on the results of the evaluation, the licensee concluded that there is sufficient margin in the original design to ensure the concrete portion of the TB structure will remain intact during and following an SSE. The NRC staff concurs with the licensee's conclusion.

3.2.2.3 Main Turbine Condenser

The main turbine condenser is a twin-shell, horizontal tube unit. The condenser shell units are massive structures, with 7/8-inch thick steel walls, that contain substantial internal bracing and are seismically rugged. Each of the two shell units of the main turbine condenser is approximately 40 ft X 30 ft X 48 ft high. The base of each condenser shell unit is rigidly mounted to the reinforced concrete TB base mat which is 26 feet below grade.

An evaluation of the seismic ruggedness of condensers and condenser anchorage for GE BWR plants is reported in Reference 6. The configurations of the GE BWR condensers were compared to condensers in the earthquake experience data. Condensers in the earthquake experience data exhibited substantial seismic ruggedness even when they were not designed to resist earthquakes. Comparisons of condenser designs in GE BWR plants with those in the earthquake experience database revealed that GE plant designs are similar to those that exhibited good earthquake performance. The study concluded that a failure and significant breach of a pressure boundary in the event of a design basis SSE is highly unlikely. This conclusion was further verified by a detailed comparison of the CNS condenser configuration to condensers in the earthquake experience data. Based on the above, the NRC staff concludes there is reasonable assurance that the main turbine condenser is seismically adequate for the proposed MSIV leakage path system.

3.2.2.4 Main Turbine Condenser Anchorage

The shell units of the main turbine condenser are self-supporting structures that do not require any external support from the TB structure at any point other than the base anchorage. The base anchorage includes bolts for tension restraint, a centrally located seismic shear key, and a thrust anchor for resisting operating loads. The licensee performed a calculation to evaluate the seismic capability of the main turbine condenser anchorage for postulated SSE loading. The licensee stated that the results of the calculation for the tension and shear in the condenser shell anchorages are adequate to ensure that the condenser units will remain intact for postulated SSE loading. The NRC staff concurs with the licensee's conclusion.

3.2.2.5 MSIV Leakage Pathway (Piping)

For the MSIV leakage pathway, the licensee conducted walkdowns to compare the subject piping systems to piping systems that have experienced strong motion earthquakes and to verify the seismic adequacy of the main steam leakage path piping. This method utilizes a comparison of piping system demand versus experience-based capacity, augmented by extensive walkdowns, worst-case calculations, and documentation to assure acceptable piping spans, piping support configurations, design attributes, and the absence of known seismic vulnerabilities.

3.2.2.5.1 Comparison to Experience Data

3.2.2.5.1.1 Piping and Pipe Supports

The piping material data, size, and schedules were obtained from piping and instrument diagrams and line specifications. Table 6-1 of Reference 1 presents a summary of the various piping, sizes, schedules, diameter-to-thickness (D/t) ratios and American Society for Testing and Materials/American Society of Mechanical Engineers (ASTM/ASME) material designation for each of the walkdown packages. Table 6-2 presents a general summary of the piping systems experience data. In Table 6-3, the licensee presented a comparison of the D/t ranges of the CNS piping to the piping experience data. The CNS piping systems in the leakage path are enveloped by the piping experience data with the following exceptions:

(1) The experience data does not specifically identify the existence of 5" diameter piping.
(2) The CNS 3/4" piping has a lower bound D/t ratio of 3.4 versus 5 in the experience data.
(3) The CNS 1" piping has a lower bound D/t ratio of 4 versus 5 in the experience data.
(4) The CNS 1" piping has a lower bound D/t ratio of 5 versus 7 in the experience data.
(5) The CNS 24" piping has a lower bound D/t ratio of 20 versus 23 in the experience data.

The licensee stated that for items (2) through (5), the lower D/t ratios are due to the use of thicker wall piping which would be stronger and have higher capacity than the experience data piping and therefore is not a concern. For item (5), the exceedance is only 12 percent which is less than typical piping system fabrication tolerances. The licensee considered that this piping is adequately represented in the experience data. The 5" diameter piping in item (1), although not explicitly represented in the database, is enveloped by those of larger and smaller sizes of piping. Therefore, the licensee concluded that this piping is adequately enveloped by the experience data and the supporting analysis. The NRC staff concurs with the licensee's conclusion.

Table 6-4(a) of Reference 1 provides a summary of the allowable stress capacity for the predominant piping materials of the experience data piping. Table 6-4(b) provides a similar summary for the CNS piping. These tables demonstrate that the CNS piping in the leakage pathway is adequately represented in the piping experience data.

In addition, Table 6-5 of Reference 1 provides a summary of minimum and maximum ratios of the actual CNS vertical support spans to the suggested American National Standards Institute (ANSI) B31.1 deadweight spans, and the actual CNS lateral support spans to the suggested ANSI B31.1 spans. Table 6-6 provides suggested B31.1 deadweight support spans. Figures 6-1 through 6-4 compare the CNS vertical support span ratios and lateral-to-vertical support span ratios to the experience piping span data. These figures show that the CNS piping support spans are well represented and adequately enveloped by the piping experience data. Accordingly, the licensee's evaluation of this matter is acceptable.

3.2.2.5.1.2 Related Equipment

Other equipment in the scope of the leakage pathway review include valves, instruments, and tanks which are referred to as related equipment in this evaluation. The Seismic Qualification Utility Group (SQUG) GIP-2 methodology, documented in Reference 8, was employed to address the seismic adequacy of the equipment. GIP-2 provides a formal procedure for evaluating these classes of equipment against the earthquake experience data. The Licensee’s implementation of the GIP-2 procedure at CNS is documented separately in Reference 9.

In Reference 1, the licensee compared the CNS SSE ground spectrum to the GIP-2 Reference Spectrum and GIP-2 Bounding Spectrum, and found that the CNS spectrum is well bounded by the GIP-2 spectrum. The NRC staff has reviewed this information and finds the licensee's evaluation acceptable.

3.2.2.5.2 Analytical Evaluations

3.2.2.5.2.1 Piping

The majority of piping systems under review were originally designed to the 1967 B31.1, “Power Piping Code.” The original design only considered loadings due to pressure, dead load, design mechanical loads, and thermal loads. The capacity criteria that were used for piping system limited analytical reviews and detailed analyses, which included consideration of a design basis SSE, are provided in References 1, 2, and 3.

The licensee performed simplified and detailed piping analysis which were conducted for selected systems in the MSIV leakage path. Detailed response spectra modal analysis was conducted for several piping systems. The analysis was a realistic seismic analysis with intermodal and inter spatial combinations by the square root of the sum of squares method. In addition to the seismic analysis, deadweight and normal operating thermal analyses were conducted.

Three piping systems were selected for detailed computer analysis: (1) Main Steam System (including the By-Pass piping), (2) Primary Leakage Pathway, and (3) Alternative Leakage Pathway. These systems were selected because they are the primary mechanisms for the delivery of the MSIV leakage to the condenser.

In addition, localized equivalent static analyses were used to (1) evaluate the effects of seismic anchor motions, (2) evaluate spatial interaction conditions, (3) evaluate localized areas of seismic vulnerability, and (4) determine loads used in the detailed support evaluations.

The licensee stated in References 1 and 3 that this analytical methodology was consistent with the approach accepted in the staff’s safety evaluation (Reference 7) for GE Topical Report, NEDC-31858P-A (Reference 6) for the Monticello Staff’s MSIV leakage pathway submittal. In addition, the licensee stated that the analytical results indicate that the piping systems at CNS have adequate capacity to maintain leak-tight structural integrity during and after a postulated SSE. The NRC staff has reviewed the above information and finds it to be acceptable.

3.2.2.5.2.2 Pipe Supports

Detailed support qualifications were conducted for all supports associated with the piping analysis described in Section 3.2.2.5.2.1, Piping, of this SE. The licensee stated that the selection of supports for detailed support qualifications was based on identifying or establishing worst case supports during the walkdowns. The basis for the determination of these worst case supports included consideration of the following: (1) short, fixed, or hard spot rod hangers that were judged to be susceptible to fatigue failure during a design basis SSE event; (2) U-bolts susceptible to significant lateral loads; (3) supports that were judged to be most susceptible to failure during a design basis seismic event based on field review; and (4) supports judged to have no ductile anchorages.

The evaluation criteria for welded and bolted anchorages were the same as those provided in the American Institute of Steel Construction (AISC) Steel Construction Manual except the allowable stress was increased by a factor of 1.7 when an SSE load was included. The GIP-2 criteria for concrete expansion anchor bolts developed by SQUG were used for the evaluation of the adequacy of concrete expansion anchor bolts. These criteria are acceptable to the NRC staff as discussed in References 7 and 8.

Table 6-10 provides a summary of the number of supports subjected to detailed analytical reviews and the basis of these reviews. These supports represent over 30 percent of the support population in the MSIV leakage path. From this review, the licensee concluded that it is reasonable to assume that the supports for the MSIV leakage path piping have adequate seismic capacity. The NRC staff concurs with the licensee's conclusion.

3.2.2.5.2.3 Related Equipment

The licensee used the GIP-2 methodology described in Reference 8. Seismic capacity provisions, anchorage, and seismic spatial interaction concerns were reviewed. The GIP-2 bounding spectrum that was obtained from earthquake experience data was used to establish the seismic capacity of all related equipment.

The licensee stated that the majority of the related equipment are valves located at the lower elevations, and valve operability is not a concern for CNS because the applicable valves are either not required to reposition to establish the leakage path, or fail safe with respect to the leakage path. The NRC staff concurs with the licensee's determination.

3.2.3 Conclusion Regarding MSIV Leakage Pathway Ruggedness

Based on the above evaluation, the NRC staff concludes that the proposed evaluation methodology identified in References 1, 2, 3, and 4 is acceptable for use in determining the seismic adequacy of the alternate leakage path, i.e., the main steam line piping and connecting components, from the MSIVs to the main turbine condenser and the TB at CNS. The NRC staff's conclusion is based on the following considerations: (1) the design attributes of the CNS main condenser are generally enveloped by those of the condensers in the earthquake experience database, and the condenser assembly has sufficient anchorage capacity, (2) the non-seismically analyzed leakage path pipes are represented by those in the earthquake experience database that demonstrated good seismic performance, (3) the detailed analyses and the worst case analysis performed for the non-seismic portion of the main steam drain lines indicated adequate safety margins for piping stresses and support loads, (4) the criteria and methodologies used for the piping analyses have been previously reviewed and accepted by the NRC for other plants, and (5) the TB has been adequately designed to withstand the SSE loads.

The NRC staff's acceptance of the methodology for this application is not an endorsement for the use of the experience-based methodology for other applications at CNS. In addition, it should be noted that the NRC staff's review did not include the new FRS that were developed by the licensee for the TB (Reference 3); the adequacy of the FRS is to be addressed by the licensee under 10 CFR Section 50.59.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding *(66 FR 48289,* dated September 19, 2001*)*. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 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) 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.

(3) Letter, Nebraska Public Power District to U.S. NRC, “Response to Request for Additional Information Related to Nebraska Public Power District’s Seismic Reevaluation Proposed to Address Cooper Nuclear Station License Condition 2.C.(6), Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46,” dated September 27, 2002.

(4) a. Letter, Nebraska Public Power District to U.S. NRC, “Response to Draft Request for Additional Information on the Supplemental Information submitted by Nebraska Public Power District for Cooper Nuclear Station License Condition 2.C.(6), Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46,” dated November 25, 2002

b. Letter, Nebraska Public Power District to U.S. NRC, "Design Basis Accident Radiological Assessment Calculation Methodology-Additional Information, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46," dated November 25, 2002.

(5) Cooper Nuclear Station Updated Safety Analysis Report.

(6) GE Topical Report, NEDC-31858P-A, Revision 2, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," August, 1999.

(7) Letter from NRC to Northern States Power Company (NSP), “Safety Evaluation on Seismic Verification of the MSIV Leakage Path at Monticello (TAC No. 96238),” September 16, 1998.

(8) “Supplemental Safety Evaluation Report No. 2 on Seismic Qualification Utility Group’s Generic Implementation Procedure, Revision 2, Corrected February 14, 1992.”

(9) “Cooper Nuclear Station Verification of Seismic Adequacy of Mechanical and Electrical Equipment, Unresolved Safety Issue A-46 (SQUG),” Nebraska Public Power District.

Attachments: Table 1
Table 2
Table 3
Table 4

Principal Contributors: Michelle Hart
Leta Brown
Yong Kim

Date: February 21, 2003

Table 1

CNS Main Steam Line Break Accident Analysis Parameters Used by the NRC Staff Source Term

TS 3.4.6 Limit, RCS Equilibrium Specific Activity, µCi/gm Dose Equivalent I-131	0.2
TS 3.4.6 Limit, RCS Maximum Specific Activity, µCi/gm Dose Equivalent I-131	4.0
Operational RCS Iodine Concentration, µCi/mL	
I-131	6.8E-2*
I-132	3.8E-1
I-133	4.0E-1
I-134	5.4E-1
I-135	5.2E-1
Coolant Release Mass, lbm	
Steam	20,000
Liquid	120,000
Coolant Release Duration (MSIV Closure), seconds	10.5

Control Room

Unfiltered inleakage (duration of the accident), scfm	
Infiltration	71
Ingress/Egress	10
Time to isolate air intake, seconds	60
Air intake rate, scfm	
0 - 1 min: Normal supply	3235
1 min - 720 hours : Emergency supply	810 = 900 -10%
Recirculation flow rate, cfm	0
Breathing rate, (duration of accident), m^3/s	3.47E-4
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
Control room proper volume, ft^3	64,640
Control room envelope volume, ft^3	141,860

Other Parameters

Dose conversion factors	FGR11/FGR12
Offsite breathing rate, offsite, m^3/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4

* Note: 6.8E-2 = 6.8×10^{-2} = 0.068

Table 2

CNS Control Rod Drop Accident Analysis Parameters
Source Term

Reactor power (2381 x 1.02), MWt	2429
Core inventory based on GE14 Amendment 22, ORIGEN2 and BWRUE	
Maximum radial peaking factor	2.0
Rods per assembly	
8x8 NB (GE9B)	60
10 x 10 (GE14)	87.3
Number of assemblies in core	548
Number of rods that fail	
8x8 NB (GE9B)	850
10 x 10 (GE14)	1200
Mass fraction of fuel in damaged rods that melts	0.0077

Control Room

Unfiltered inleakage (duration of the accident), scfm	
Infiltration	71
Ingress/Egress	10
Time to isolate air intake, hours	24
Air intake rate, scfm	
0 - 24 hours: Normal supply	3235
24 - 720 hours : Emergency supply	810 = 900 -10%
Recirculation flow rate, cfm	0
Breathing rate, (duration of accident), m^3/s	3.47E-4
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
Control room proper volume, ft^3	64,640
Control room envelope volume, ft^3	141,860

Other Parameters

Dose conversion factors	FGR11/FGR12
Offsite breathing rate, offsite, m^3/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
Atmospheric dispersion factors	Table 3

Table 3

CNS Atmospheric Relative Concentration (X/Q) Values

The NRC staff finds use of the following exclusion area boundary (EAB), and low-population zone (LPZ) X/Q values (sec/m^3) acceptable for postulated ground level releases from the RB and TB vent and the elevated plant stack. These values were previously found acceptable in CNS Amendments 183 and 187.

Receptor Location	Ground level X/Q	Stack X/Q
EAB		
0 - 90 sec	5.2 E-4	
90 sec - 0.5 hrs		1.2 E-4*
0.5 - 2 hrs		1.6 E-5
LPZ		
0 - 90 sec	2.9 E-4	
90 sec - 0.5 hrs		1.4 E-4*
0.5 - 8 hrs		4.0 E-5
8 - 24 hrs		1.6 E-5
1 - 4 days		5.8 E-6
4 - 30 days		1.7 E-6

The NRC staff finds the following control room X/Q values (sec/m^3) acceptable for postulated ground level releases from the RB and TB, and the elevated plant stack release. These values were previously found acceptable in CNS Amendments 183 and 187, with the exception of the TB releases. These were granted interim approval at that time, but are found acceptable for use at CNS with this amendment.

Receptor Location	Reactor Bldg. X/Q	Turbine Bldg. X/Q	Stack X/Q
Control Room			
3.8 - 10 secs	3.77 E-3		
10 - 90 secs	4.07 E-3		
90 secs - 0.5 hr			3.03 E-4*
0.5 - 2 hrs			1.00 E-9
0 - 2 hrs		9.54E-4	
2 - 8 hrs		4.93E-4	2.65 E-9
8 - 24 hrs		2.69E-4	6.41 E-8
1 - 4 days		1.72E-4	2.00 E-8
4 - 30 days		1.43E-4	1.66 E-8

* The LOCA ECCS release assumes only a stack release with fumigation for 0 to 0.5 hours.

Table 4

CNS Calculated Radiological Consequence Analysis Results

Offsite Dose

	EAB (rem)	LPZ (rem)	Acceptance Criterion (rem)
CRDA			
Thyroid	0.78	2.03	75
Whole Body	0.10	0.11	6.25
MSLB, equilibrium iodine concentration			
Thyroid	0.538	0.32	30
Whole Body	0.0072	0.00429	2.5
MSLB, maximum iodine concentration			
Thyroid	10.76	6.4	300
Whole Body	0.144	0.0858	25

Control Room Dose

	CNS Control Room (rem)	Acceptance Criterion (rem)
CRDA		
Thyroid	6.44	30
Whole Body	0.008	5
MSLB, equilibrium iodine concentration		
Thyroid	5.77	30
Whole Body	0.00155	5

Cooper Nuclear Station

cc:

Mr. William J. Fehrman
President and Chief Executive Officer
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

Mr. Michael T. Coyle
Site Vice President
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

Mr. John R. McPhail, General Counsel
Nebraska Public Power District
P. O. Box 499
Columbus, NE 68602-0499

Mr. Paul V. Fleming, Licensing Manager
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

Mr. Michael J. Linder, Director
Nebraska Department of Environmental
Quality
P. O. Box 98922
Lincoln, NE 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, NE 68305

Ms. Cheryl K. Rogers, Program Manager
Nebraska Health & Human Services
System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P. O. Box 95007
Lincoln, NE 68509-5007

Mr. Ronald A. Kucera, Director
of Intergovernmental Cooperation
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65102

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 218
Brownville, NE 68321

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Jerry Uhlmann, Director
State Emergency Management Agency
P. O. Box 116
Jefferson City, MO 65101

Chief, Radiation Control Program, RCP
Kansas Department of Health
and Environment
Bureau of Air and Radiation
1000 SW Jackson
Suite 310
Topeka, KS 66612-1366

Mr. Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
401 SW 7th Street
Suite D
Des Moines, IA 50309

Mr. Scott Clardy, Director
Section for Environmental Public Health
P.O. Box 570
Jefferson City, MO 65102-0570