April 7, 2000

Mr. J. H. Swailes
Vice President of Nuclear Energy
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT ON DESIGN-BASIS ACCIDENT RADIOLOGICAL ASSESSMENT CALCULATIONAL METHODOLOGY REVISION (TAC NO. MA7758)

Dear Mr. Swailes:

The Commission has issued the enclosed Amendment No. 183 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The application requested changes to CNS design-basis accident radiological assessment calculational methodology as described in your submittal dated December 22, 1999, as supplemented by letters dated March 20, March 24 (2), March 29, and April 5, 2000. Because of the changes in methodology and resulting dose assessments consequences, Nebraska Public Power District (NPPD) considered these revisions to represent an unreviewed safety question under 10 CFR 50.59 and the requested changes were made pursuant to 10 CFR 50.90.

The amendment approves, as described in the enclosed safety evaluation, revisions to the radiological assessment calculational methodology for the loss-of-coolant accident (LOCA) and the control rod drop accident (CRDA). However, the staff is deferring the review of implementing this change on a permanent basis. Subsequently, this amendment is effective immediately and shall remain effective until CNS enters mode 4 in preparation for refueling outage 20 (effectively, one operating cycle). Also, the staff has deferred review of the radiological assessment methodology revisions for the fuel handling accident (FHA) and the main steamline break (MSLB) accident. It is anticipated that NPPD will resolve any outstanding issues concerning these calculational methodology revisions in a timely manner in support of a permanent change that is acceptable to the staff.

One of the outstanding differences in the assumptions for calculating doses to control room personnel is the consideration of fumigation conditions with respect to the elevated release path for the first 30 minutes following a postulated LOCA. The staff's position is that CNS's licensing basis with respect to this issue is derived from NPPD's letter dated December 30, 1980, regarding Post-TMI Requirements/Action Plan. In accordance with guidance available at the time, NPPD incorporated fumigation conditions for the first 30 minutes after an accident. Due to the scope and complexity of the review, and in order to facilitate plant startup, the staff offered to expedite review of this issue provided that the current licensing basis remain unchanged. In response, NPPD has provided an evaluation of the effects of assuming an additional dose contribution from the elevated release path by assuming an initial 30-minute fumigation contribution. Based upon the evaluation and due to the difference in positions between the staff and NPPD concerning the inclusion of fumigation conditions, NPPD has agreed to continue its commitment to implement a procedure to provide control room personnel with potassium iodide (KI) thyroid-blocking tablets upon indications of a LOCA that results in core damage. NPPD's commitment regarding the use of KI provides an acceptable basis for approving the revisions to the calculational methodology. It is the staff's understanding that

NPPD intends to revisit the issue of including fumigation conditions for the first 30 minutes following an accident and the associated commitment involving KI. It is expected that NPPD will provide sufficient information to resolve this issue of including fumigation conditions by refueling outage 20.

An evaluation of the seismic adequacy of the main steam piping, main turbine condenser, and turbine building is integral to crediting iodine removal by “plate-out” in the main turbine condenser after a postulated accident. NPPD has provided sufficient information to justify the operability of the main steam piping and the main turbine condenser following a safe shutdown earthquake (SSE) so that the iodine removal can be accomplished. However, a more technically detailed analysis is required to justify full qualification which will ensure long-term acceptability. NPPD has committed to provide this evaluation in a timely manner. The staff believes that this commitment, which is relied upon by the staff to approve this licensing action, is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, the license is amended by the addition of condition 2.C.(6).

Based on the information provided by NPPD regarding the LOCA and CRDA analyses, the results of the staff’s confirmatory calculations, NPPD’s continuing commitment to provide KI to control room personnel, the staff finds reasonable assurance that, for the effective duration of this amendment, the postulated radiological consequences of the design-basis LOCA and CRDA at CNS will be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, General Design Criterion 19, and Sections 6.4 (LOCA and CRDA) and 15.4.9 (CRDA) of NUREG-0800. Therefore, the changes to the LOCA and CRDA calculational methodologies, as described in the enclosed safety evaluation, are acceptable on an interim basis.

Proper planning regarding submission of licensing requests is necessary to allow sufficient allocation of NRC and NPPD resources. NPPD submitted this request on December 22, 1999, and requested an issuance date that would support CNS startup in early April 2000. In light of the scope and complexity of the requested changes, and the importance of the issue, the staff feels that sufficient preparation and planning was not given to this issue by NPPD.

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/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 183 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

NPPD intends to revisit the issue of including fumigation conditions for the first 30 minutes following an accident and the associated commitment involving KI. It is expected that NPPD will provide sufficient information to resolve this issue of including fumigation conditions by refueling outage 20.

An evaluation of the seismic adequacy of the main steam piping, main turbine condenser, and turbine building is integral to crediting iodine removal by “plate-out” in the main turbine condenser after a postulated accident. NPPD has provided sufficient information to justify the operability of the main steam piping and the main turbine condenser following a safe shutdown earthquake (SSE) so that the iodine removal can be accomplished. However, a more technically detailed analysis is required to justify full qualification which will ensure long-term acceptability. NPPD has committed to provide this evaluation in a timely manner. The staff believes that this commitment, which is relied upon by the staff to approve this licensing action, is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, the license is amended by the addition of condition 2.C.(6).

Based on the information provided by NPPD regarding the LOCA and CRDA analyses, the results of the staff’s confirmatory calculations, NPPD’s continuing commitment to provide KI to control room personnel, the staff finds reasonable assurance that, for the effective duration of this amendment, the postulated radiological consequences of the design-basis LOCA and CRDA at CNS will be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, General Design Criterion 19, and Sections 6.4 (LOCA and CRDA) and 15.4.9 (CRDA) of NUREG-0800. Therefore, the changes to the LOCA and CRDA calculational methodologies, as described in the enclosed safety evaluation, are acceptable on an interim basis.

Proper planning regarding submission of licensing requests is necessary to allow sufficient allocation of NRC and NPPD resources. NPPD submitted this request on December 22, 1999, and requested an issuance date that would support CNS startup in early April 2000. In light of the scope and complexity of the requested changes, and the importance of the issue, the staff feels that sufficient preparation and planning was not given to this issue by NPPD.

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/
Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 183 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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Cooper Nuclear Station

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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
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 December 22, 1999, as supplemented by letters dated March 20, March 24 (2), March 29, and April 5, 2000, 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, Facility Operating License No. DPR-46 is hereby amended as indicated in the attachment to the 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 and shall remain effective until CNS enters mode 4 in preparation for refueling outage 20.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment: Changes to the Operating License

Date of Issuance: April 7, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 183

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of Operating License DPR-46 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
4	4
--	5

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) No later than 8 weeks after the Cooper Nuclear Station (CNS) Cycle 21 startup, the licensee shall submit a request for the staff to review and approve a seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building. The evaluation will be performed to assess the ability of the aforementioned main steam piping and main turbine condenser to remain sufficiently intact to direct main steam leakage from the MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design-basis accident dose calculations during and after a Safe Shutdown Earthquake. This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation. The licensee's approved request shall be fully implemented, including the completion of modifications, within 12 months of approval or prior to CNS Cycle 22 startup, whichever is later.

D. This license is subject to the additional following conditions for the protection of the environment:

The licensee shall, for operation not later than April 30, 1975, modify the liquid and gaseous radiological effluent handling systems in accordance with the systems described in Amendment 18 to the Final Safety Analysis Report. If such systems are not installed by such date, the licensee shall, nonetheless, observe the gaseous activity release limits set forth in paragraph a.4 of Section 2.4.3 of Appendix B attached hereto and facility operation shall be restricted accordingly, is necessary.

E. This license is effective as of the date of issuance and shall expire at midnight, January 18, 2014.

FOR THE ATOMIC ENERGY COMMISSION

ORIGINAL SIGNED BY:

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachments:
Appendices A & B - Technical Specifications
Appendix C - Additional Conditions

Date of Issuance: January 18, 1974

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-46

REGARDING CHANGES TO DESIGN BASIS ACCIDENT RADIOLOGICAL

ASSESSMENT CALCULATIONAL METHODOLOGY

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated December 22, 1999, as supplemented by letters dated March 20 and 24 (2), March 29, and April 5, 2000, the Nebraska Public Power District (NPPD) submitted a request for amendment to License No. DPR-46 to revise the Cooper Nuclear Station (CNS) design-basis accident (DBA) radiological assessment calculation methodology. The reasons for these changes are to incorporate more recent site-specific meteorological data and a revised methodology for calculating relative concentration values (X/Q), to reflect plant-specific system operating parameters and design, to utilize more widely accepted assumptions, to incorporate the Technical Information Document (TID-14844) source term to be consistent with the accident assumptions used, and to allow new fuel parameter considerations to include higher burnup fuel designs. Because of the changes in methodology and resulting dose assessments consequences, NPPD considered these revisions to represent an unreviewed safety question under 10 CFR 50.59 and the requested changes were made pursuant to 10 CFR 50.90. The March 20 and 24 (2), March 29, and April 5, 2000, letters provided additional clarifying information that was within the scope of the original application and *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

The December 22, 1999, licensing amendment request provided a substantial change from the previous licensing bases assumptions and methodology for four DBA analyses. These DBAs included the loss-of-coolant accident (LOCA), control rod drop accident (CRDA), fuel handling accident (FHA), and main steamline break (MSLB) accident. Due to the scope and complexity of the changes requested, and in order to facilitate plant startup after refueling outage 19 (RFO-19), NPPD stated, in its March 24, 2000 letter, that it would accept deferral of the review of the FHA and the MSLB accident. Until review and approval for the FHA analysis is completed, NPPD will impose limitations, as described in NPPD's letter dated March 24, 2000, on the movement of any irradiated GE14 fuel assemblies or loads over irradiated GE14 fuel assemblies. With respect to the MSLB analysis, NPPD states that the source term associated with the MSLB accident is limited by the technical specification addressing reactor coolant system specific activity. Therefore, this evaluation is limited to the portions of NPPD's request related to the LOCA and CRDA.

As a result of requests for additional information addressing the technical basis for design inputs and assumptions, NPPD reanalyzed the radiological consequences of the proposed changes. This reanalysis was submitted by letters dated March 24, 2000 (2).

With respect to previous licensing actions that are related to the staff's review of this request, during review and approval of Amendment 167, dated January 27, 1995, NPPD committed to implement a procedure to provide control room operators with potassium iodide (KI) throid-blocking tablets upon indications of a LOCA that results in core damage. NPPD, as stated in its letters dated March 24 and April 5, 2000, will continue this commitment until unresolved issues surrounding the use of ARCON96 stack release methodology are resolved.

2.0 EVALUATION

As stated above, this evaluation is limited to the LOCA and CRDA analyses. In order to support startup from RFO-19, the review of the FHA and MSLB are deferred because (1) approval of the FHA analysis, as revised in the December 22, 1999, letter is only needed to support movement of irradiated GE14 fuel or loads over irradiated GE14 fuel assemblies, which is not expected to be required during Cycle 20 operation (for any eventuality requiring movement of non-GE14 fuel, NPPD can continue to rely on the existing licensing basis related to the FHA analysis) and (2) approval of the MSLB accident analysis, as revised in the December 22, 1999, letter, is not needed due to compliance with the CNS technical specifications, which limits the source term associated with the MSLB accident. Therefore, the staff finds that deferral of the FHA and MSLB accident analyses is acceptable.

NPPD 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 NPPD and utilized by the staff to perform independent calculations. The results of the staff's independent calculations and evaluations were used to confirm the acceptability of the licensee's analysis methodology. Based upon comparisons of results and the considerations described below, the staff found the licensee's analysis to be acceptable.

The design inputs utilized by the staff to evaluate these accidents are given in Tables 1, 2 and 3. During the review of these accidents, the staff and NPPD discussed several issues that warranted the use of compensatory measures or the limitation of this amendment to one fuel cycle in order for the staff to find the proposed request acceptable. These issues include the design-basis source term, the main steam isolation valve (MSIV) leakage pathway, and the meteorology utilized. These areas are discussed in detail below.

2.1 Design-Basis Source Term

NPPD determined that inclusion of higher burnup fuel design considerations (i.e., GE14 fuel design) into the CNS DBA radiological assessment should be conducted concurrently with the revised DBA radiological assessment methodology. This was done to support startup following RFO-19 and to ensure that fuel management could be optimized in subsequent fuel operating cycles. The source term methodology proposed in the CNS submittal utilizes a source term from TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

In a letter dated March 6, 2000, the staff requested NPPD to assure that this source term was conservative with respect to the limiting design parameters of the fuels to be used (i.e., GE14 fuel). The staff noted that although the nuclides of interest in design-bases accident analyses typically reach equilibrium values early in a fuel cycle, extended burnups can affect the core inventory. A substantial fraction of the energy produced during the final fuel irradiation cycle may be derived from plutonium-239 (Pu-239). The most significant difference in terms of radiological analyses is approximately 27 percent greater iodine-131 (I-131) yield from Pu-239 fissions as compared with that for uranium-235 (U-235) fissions. The staff also noted that TID-14844 values were based upon a simplified formula that did not consider nuclide ingrowth. In a letter dated March 20, 2000, NPPD stated that the source term used is based on TID-14844 values and methodologies, with extended fuel burnup corrections applied in accordance with NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors." NPPD also stated that the extended burnup fuel designs utilized at CNS (including GE14) are within the applicability of NUREG/CR-5009 and that no additional modification of the source term is necessary to accommodate the most limiting fuel design. The staff is not convinced, at this time, that the source term taken from TID-14844 is appropriate for the GE14 fuel design for extended burnup conditions and believes that further review is necessary.

However, the staff believes, based upon previous experience, that the inventory of Pu-239 will be small enough that the impacts on the source term will be minimal or negligible for the first cycle of extended burnup fuel. Therefore, there is reasonable assurance that the TID-14844 source term will provide a bounding source term for approximately one operating cycle (until CNS enters mode 4 in preparation for refueling outage 20). Subsequently, the use of the TID-14844 source term, as submitted, is acceptable until CNS enters mode 4 in preparation for RFO-20 (effectively one operating cycle). It is expected that NPPD will provide sufficient information to resolve this issue to support a permanent change to the radiological assessment calculational methodology by RFO-20.

2.2 MSIV Leakage Pathway Analysis

The main steam system transports steam from the reactor vessel to the main turbines and to other steam-driven auxiliary equipment. Each main steam line, of which there are four, has two air-operated MSIVs, one inboard and one outboard of the steamline containment penetration. In the event of a LOCA, the MSIVs close to isolate the reactor to prevent the direct release of fission products from the reactor to the environment. Due to the size and service conditions of the MSIVs, the MSIVs are not leak tight. Current technical specifications specify an allowable leakage rate of 11.5 standard cubic feet per hour for each steam line.

NPPD analyzed the radiological consequences of the MSIV leakage separately from the containment leakage. NPPD then added the contribution of each pathway (along with the postulated doses for the emergency core cooling system (ECCS) leakage pathway) to obtain the total doses for comparison to the 10 CFR Part 100 dose guidelines and the General Design Criterion 19 of 10 CFR Part 50 (GDC 19) control room criteria. The staff found the licensee's doses from containment leakage and ECCS leakage to be acceptable. NPPD analyzed the MSIV leakage pathway assuming credit for partitioning and plateout in the turbine and main turbine condenser. In its request for additional information, the staff requested NPPD to justify this methodology in light of seismic considerations. NPPD has provided information to justify the operability of the main steam piping, the main turbine condenser, and the turbine building, in the near term, following a safe shutdown earthquake (SSE) (see Section 3.0 for discussion); therefore, iodine removal by partitioning and plateout can be credited.

An evaluation of the seismic adequacy of the main steam piping, main turbine condenser, and turbine building is integral to crediting iodine removal by "plate-out" in the main condenser after a postulated accident. NPPD has provided sufficient information to justify the operability of the main steam piping and the main turbine condenser following an SSE so that the iodine removal can be accomplished. However, full qualification, which involves completion of a more technically detailed analysis, is required to ensure long-term acceptability. NPPD has committed to provide this evaluation in a timely manner. The staff believes that this commitment, which is relied upon by the staff to approve this licensing action, is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, the license is amended by the addition of Condition 2.C.(6).

2.3 <u>Meteorology Considerations</u>

NPPD calculated X/Q values for the exclusion area boundary (EAB) and low population zone (LPZ) using site-specific inputs and the methodology described in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." Calculations were made for an EAB distance of 920 meters and LPZ distance of 1609 meters. Releases from the turbine building were assumed to be ground level. Building wake corrections were applied for the 0- to 8-hour time period using a minimum building cross-sectional area of 1569 square meters. Releases from the 99.1-meter stack were calculated as elevated, with fumigation conditions assumed to occur during the first 30 minutes of the accident. The calculated effective height of the elevated release also factored in changes in the maximum terrain height in the site vicinity.

NPPD used the ARCON96 methodology described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake," with several modifications, to calculate X/Q values for the control room dose assessment, other than for fumigation conditions. The modifications resulted from discussions with the NRC staff during its review of the licensee's December 22, 1999, submittal. NPPD subsequently provided modified X/Q calculations by letter dated March 24, 2000.

For calculations using the ARCON96 methodology NPPD used onsite meteorological data collected during calendar years 1994 through 1998. In 1995 and 1996, data recovery of temperature difference measurements involving the 10-meter level was less than the recommended minimum of 90 percent cited in Regulatory Guide 1.23, "Onsite Meteorological Programs." NPPD also noted that instrument accuracy limits for differential temperatures in excess of 5.28 degrees Celsius per 100 meters may be outside of the recommendations of Regulatory Guide 1.23, but values of this magnitude are beyond those given in Regulatory Guide 1.23 and therefore would not affect determination of the atmospheric stability category. Other than these exceptions, NPPD confirmed that overall the meteorological measurement program met the guidance provided in Regulatory Guide 1.23. When using the ARCON96 methodology, the licensee performed X/Q calculations for each of the 5 years individually. The highest X/Q for each time period, regardless of year, was selected for input into the dose calculations. Use of the highest X/Q results in the highest dose estimate when compared with using a lower X/Q calculated for any of the other years. Only one of the highest X/Q values occurred when data recovery was less than 90 percent. For this exception, the other years having at least 90 percent data recovery resulted in lower X/Qs. Thus, these exceptions do not appear to be of significance. The staff noticed occasional occurrences of very unstable conditions during the night. NPPD attributed this phenomena to factors such as wind shifts and minor temperature fluctuations.

Using the ARCON96 methodology, NPPD performed X/Q calculations assuming a diffuse release from the turbine building based on a loss of offsite power (LOOP). The licensee also performed a calculation assuming no LOOP with a point release from a common exhaust located further away from the control room than the turbine building. Following discussions with the staff, the licensee modified the diffuse release calculation to divide both the assumed height and width of the area of release by a factor of 6 when estimating the initial diffusion coefficients. Since the assumption of LOOP resulted in a higher postulated dose, the licensee used the LOOP X/Q in the dose assessment to demonstrate compliance with GDC 19.

Initial estimates made by the licensee for postulated releases from the stack also used the ARCON96 methodology. However, this methodology does not consider fumigation, for the first 30 minutes following an accident, which is part of the CNS design basis.[1] Due to the scope and complexity of the review, and in order to facilitate plant startup, the staff offered to review this issue provided that the current licensing basis remain unchanged. In response, CNS provided an evaluation by letter dated March 24, 2000, which included the effects of fumigation for the first 30 minutes of an accident. This control room X/Q for fumigation was applied for the first 30 minutes of the accident and is the same value that was provided previously in the CNS Post-TMI Requirements/Action Plan. The March 24, 2000, letter estimated the LOCA doses using the fumigation X/Q as 42 rem. The NPPD evaluation did not apply any credit for mitigation of this dose by utilizing KI, although NPPD, in letters dated March 24 and April 5, 2000, committed to continue an interim compensatory measure to provide reasonable assurance that GDC 19 limits are met. This interim compensatory measure is to continue implementation of the commitment to make KI tablets available to the control room personnel (in accordance with recommended dosage) if plant conditions indicate that a LOCA is occurring coincident with core damage. Using the CNS fumigation X/Q, the LOCA analysis parameters provided by CNS (Table 1), and applying a protection factor of 10 for utilization of KI, the staff performed its own evaluation of the control room operator doses. Based upon comparison of the CNS evaluation and the staff's results, the staff agrees that by utilizing KI, CNS can meet GDC 19 requirements.

It is the staff's understanding that NPPD may revisit the issue of including fumigation conditions for the first 30 minutes following an accident and the associated commitment involving KI.

2.4 Control Room Habitability Generic Issue

The 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. Recent testing by 20 licensees using enhanced test methods have shown that in all 20 cases, the measured infiltration rates exceeded the values

[1] CNS previously accounted for fumigation as part of the Control Room Habitability review of item III.D.3.4 attached to the CNS post-TMI requirements action plan letter dated December 30, 1980.

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 those facilities that have not performed the enhanced testing. The staff is currently participating in an NRC-industry initiative to resolve these concerns.

The staff has determined that there is reasonable assurance that the CNS control room will be habitable during design-basis accidents and that this amendment may be approved before the resolution of this generic issue. The staff bases this determination on the availability of KI as an interim compensatory measure. The approval of this amendment does not exempt NPPD from regulatory actions that may be imposed in the future as this generic issue is resolved.

3.0 EVALUATION OF THE STRUCTURAL INTEGRITY OF MAIN STEAM PIPING AND MAIN TURBINE CONDENSER FOLLOWING AN SSE

3.1 Staff Approach

When crediting dose consequence mitigation due to iodine plateout in the main turbine condenser, the staff evaluated the ability of the main steam lines, main turbine condenser, and the turbine building (TB) to remain structurally intact following an SSE. Full qualification for the purpose of dose consequence mitigation would be demonstrated by satisfactory completion of a technically detailed seismic evaluation (that is acceptable to the staff) of the ability of these SSCs to maintain sufficient structural integrity during and after an SSE. An acceptable method for providing full qualification could follow the pertinent guidelines contained in the staff's safety evaluation dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Sytems,' September 1993." As full qualification has not been accomplished (i.e., this more technically detailed seismic evaluation is not available), iodine plateout in the main turbine condenser may be credited, in the near term, by demonstrating the operability of these SSCs following an SSE. The main steam lines and main turbine condenser are considered operable when NPPD utilizes engineering judgement that is supported by simplified calculations (in contrast to a more technically detailed analysis for full qualification) to demonstrate the ability of these SSCs to direct MSIV leakage to the main turbine condenser. Once operability is demonstrated for the short term, the establishment of full qualification of the SSCs is ultimately necessary to provide greater assurance that these SSCs will remain structurally intact following an SSE for the purpose of dose consequence mitigation for the long term.

3.2 Introduction

In justifying the iodine plateout in the condenser (as described in Section 2.2), NPPD provided a summary, dated March 24, 2000, as supplemented by letter dated March 29, 2000, of the seismic and structural design of the main turbine condenser, the MSIV leakage pathway (piping) to the main turbine condenser, and the turbine building (TB) structure. This information is intended to establish the operability of these components and structure following an SSE.

The staff believes that justification of the capability to direct MSIV leakage from the MSIVs to the main turbine condenser is necessary for crediting dose consequence mitigation by iodine plateout on the condenser. An acceptable method for providing full qualification could follow the pertinent guidelines contained in the staff's safety evaluation dated March 3, 1999, "Safety

Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993." The staff recognizes that the technical nature of this evaluation and the identification of plant modifications that may be necessary to support this evaluation may require significant NPPD resources. The staff believes that this justification is necessary to support the long-term acceptability of the main turbine condenser and MSIV leakage pathway (piping) to perform the dose consequence mitigation function. In a letter dated March 24, 2000, NPPD committed to provide this evaluation in a timely manner. The staff believes that this commitment is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, License No. DPR-46 is amended with additional condition 2.(C).(6) (See section 3.5). Absent this long-term evaluation, and using engineering judgement (supported by simplified calculations) along with the guidance of Generic Letter 91-18, NPPD, by letter dated March 24, 2000, submitted sufficient information to demonstrate the operability of the main steamline piping exiting from the MSIVs to the main turbine condenser, the main turbine condenser, and the TB in the event of an SSE (as described below).

3.3 Turbine Building

In a letter dated March 24, 2000, as supplemented by letter dated March 29, 2000, NPPD stated that the TB houses the main turbine condenser and a majority of the primary MSIV leakage pathway (piping) from the MSIVs to the main turbine condenser (some of the piping is located in the Class I reactor building steam tunnel). The TB base mat is reinforced concrete. The TB is a reinforced concrete structure up to the operating floor. Structural steel framing (superstructure) rises above the operating floor. The interior walls of the TB are reinforced concrete, with concrete block enclosing smaller areas. The TB was designed to the requirements for Class II structures, systems, and components (SSCs), including 100-mph wind loading and 0.1g Uniform Building Code (UBC) seismic loading.

The TB superstructure above the operating floor (at elevation 932 ft-6 in.) was evaluated as part of NPPD's Individual Plant Examination for External Events (IPEEE). NPPD's IPEEE report concluded that the TB superstructure was screened for a 0.3g Review Level Earthquake. The IPEEE report indicates that the seismic load computed using UBC criteria is well below the load in the transverse direction generated by the wind shear. The licensee provided reasonable assurance that the main TB superstructure will remain intact following an SSE (0.2g ground input acceleration) without gross structural failure.

In addition, NPPD has performed an evaluation of the TB concrete structure to confirm that it is capable of remaining structurally intact without gross structural failure following a postulated SSE. Samples of key TB substructures (e.g., walls, floor slabs, and columns) were evaluated for increased seismic loading resulting from a postulated SSE. The horizontal seismic acceleration input to the operating floor of the TB at elevation 932 ft-6 in. due to the TB response was assumed to be 0.3g based on a comparison with Class I structures (reactor building and control building). The evaluations show that the increase in design loadings from the original seismic Class II criteria to the postulated SSE condition do not result in stresses that exceed the allowable limits applicable to the SSE load case. Therefore, the licensee concludes that there is sufficient margin in the original design to ensure that the concrete portion of the TB structure will remain intact during and following an SSE. These results are based primarily on the fact that allowable stresses are increased for the SSE load case and, consequently, the increase in seismic loading is offset by the increase in allowable stresses.

NPPD stated in its March 24, 2000, letter, that it has reasonable assurance that the TB will remain structurally intact following an SSE without gross structural failure. Based on the seismic ruggedness of the TB as demonstrated by the seismic evaluation provided by NPPD, the staff concludes that the TB would remain operable in the event of an SSE.

3.4 Main Turbine Condenser

In its letter dated March 24, 2000, NPPD stated that the main turbine condenser is a twin-shell, horizontal tube unit, cooled by river water. There are two shell units of the condenser. The condenser shell units are massive structures, with 7/8-inch thick steel shell walls, that contain substantial internal bracing and are seismically rugged. The main turbine condenser is located beneath the low pressure cylinders of the main turbine. To accommodate thermal expansion, a rubber belt expansion joint is provided for each condenser neck.

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. The top of each unit is located approximately 22 feet above grade elevation. These units 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 two shell units are interconnected by a large, rounded edge, rectangular-shaped steel passageway approximately 8 ft long with cross-sectional dimensions of 14 ft-6 in. x 9 ft-6 in. This interconnection was originally field welded to the condenser shells.

NPPD has performed a calculation to evaluate the seismic capability of the main turbine condenser anchorage for postulated SSE loading. The calculation determined that seismic loading up to approximately 0.6g horizontal acceleration can be postulated before any tension in the four perimeter anchorages of a condenser shell unit would be developed from a postulated seismic event. In addition, the calculation determined that the seismic anchorage in the center of each condenser shell unit is capable of resisting a horizontal acceleration up to approximately 1g when using stress allowables for the loading condition that includes the SSE load. The maximum expected horizontal acceleration for the postulated SSE would be less than 0.3g; therefore, the calculation concludes that the existing tension and shear anchorage details for the condenser shell units are adequate to ensure that the condenser units will remain operable for postulated SSE loading.

NPPD further stated that the main turbine condenser is a seismic Class II structure/component that was originally designed for lateral seismic forces resulting from a horizontal base shear of 0.1g (UBC provisions) in combination with design operating loads (e.g., shell design pressures of 20 psig and 30" Hg vacuum). Vertical seismic loading was not included in the original design; however, the previously mentioned calculation has concluded that the vertical seismic acceleration for a postulated SSE would not have a substantial effect on the condenser shell unit anchorage. NPPD stated in its letter dated March 24, 2000, that it has reasonable assurance that the condenser will remain structurally intact following a postulated SSE. Based on the seismic ruggedness of the condenser as demonstrated by the seismic evaluation provided by NPPD, the staff considers that the main turbine condenser would remain operable during and after an SSE.

3.5 MSIV Leakage Pathway (Piping) to the Condenser

NPPD stated in its March 24, 2000, letter, that piping in the TB is designed and installed to the USAS B31.1 “Power Piping” Code (1967). In accordance with the code, the piping is analyzed for pressure, deadweight, and thermal loads. The piping is classified as Seismic Class IIS, corresponding to a seismic category Class II. In addition to the pressure and deadweight load cases, the piping is supported/restrained to withstand seismic horizontal forces equal to 0.10 times the operating dead load of the piping.

The main steam piping system downstream of the MSIV’s (beginning at the steam tunnel anchor), including the main steam piping from the turbine bypass valve to the main turbine condensers, is designed and supported/restrained to seismic Class IIS requirements. In addition to the seismic Class IIS requirements, this piping system is analyzed for dynamic loading (steam hammer) associated with a turbine stop valve closure event and for fatigue associated with normal system vibration. The results of the analyses show that the resultant pipe stresses for all load cases considered are less than the applicable Code (B31.1) allowable stress limits. The licensee has also performed a preliminary analysis of this primary pathway from the MSIVs to the main turbine condenser which has demonstrated that this piping and its associated pipe supports will remain within operability limits under postulated SSE loading.

The dynamic analysis shows that under postulated SSE loading the maximum calculated stress in the system (approximately 28,000 psi) is less than the operability limit of 36,000 psi ($2.4S_h$ or 2.4 x 15,000 psi). Amplified floor response spectra do not exist for the TB; therefore, horizontal seismic loads were computed using the ground response spectra multiplied by 1.5 with 5 percent damping. The operability limit is established based on the recommendations of Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability.” Support loads were reviewed and found to be similar in magnitude to those produced by the steam hammer event associated with the turbine stop valve closure. The system has previously experienced a steam hammer event and did not sustain any damage. This would indicate that the supports would also remain operable under SSE loading. Additionally, supports with higher loads were examined and found to be operable by engineering judgement.

In general, the staff does not accept the estimation of amplified response spectra by using the ground response spectra for a 5 percent damping multiplied by a factor of 1.5. However, in this particular case, over 95 percent of the MSIV leakage pathway (piping) and piping supports are installed within the reinforced concrete structure of the TB, as stated in NPPD’s letter dated March 29, 2000 (this provides a high confidence that the MSIV leakage pathway (piping) will remain operable following an SSE). The estimate of floor response spectra by using ground response spectra multiplied by a factor of 1.5 is considered reasonable for this application.

Based on the seismic ruggedness of the MSIV leakage pathway (piping) as demonstrated by the seismic evaluation provided by NPPD, the staff finds that the main steam piping downstream of the MSIV would remain operable in the event of an SSE.

3.6 Additional Measures for Structural Integrity Verification

NPPD further stated in its March 24, 2000, letter that the SSCs discussed in the MSIV leakage pathway including the TB are periodically inspected for a variety of reasons. For example, the TB structure and piping and equipment supports within the TB are subject to periodic structural inspections in support of the Maintenance Rule activities. The last structural Maintenance Rule inspection walkdowns were performed in 1996 and are scheduled on a 5-year cycle. The NRC has previously inspected the CNS structural Maintenance Rule program and subsequently published NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," which discusses the CNS program. Furthermore, the main steam piping system is inspected each cycle to potentially identify any deficiencies with pipe supports.

NPPD is also conducting structural integrity walkdowns of the main turbine condenser, main steam piping systems, and the TB structure during refueling outage 19 to confirm that no obvious gross structural inadequacies currently exist on these SSCs.

The licensee also stated that it plans to expand the scope of the Maintenance Rule structural inspections to include the subject SSCs as determined to be appropriate by the CNS Maintenance Rule program.

The staff finds that the licensee's additional measures to verify the structural integrity of the MSIV leakage pathway system and components, including their supports, will enhance the assurance of its seismic adequacy.

3.7 Conclusion on Structural Integrity

The staff has reviewed the licensee's submittals which provide the justification for crediting iodine removal in the main turbine condenser. The justification includes an operability demonstration of the main steamline piping exiting from the MSIVs to the main turbine condenser, the main turbine condenser, and the TB in the event of an SSE. Based on the staff's evaluations delineated above, the staff considers that the licensee's justification provides reasonable assurance that the SSCs mentioned above are seismically adequate and will be operable in the event of an SSE. The establishment of full qualification of the main steam piping and main turbine condenser to perform the dose consequence mitigation function is ultimately necessary to correct the nonconforming or degraded condition. In its letter dated March 24, 2000, NPPD made a commitment to justify full qualification by providing an assessment of the seismic adequacy of these SSCs and their supports in accordance with the milestone schedules described in the proposal. The staff believes that this commitment, which is relied upon by the staff to approve this licensing action, is of such importance to safety that no change should be made without prior staff approval in accordance with 10 CFR 50.90. Consequently, the following is added to License No. DPR-46 as additional condition 2.(C).(6):

> No later than 8 weeks after the Cooper Nuclear Station (CNS) Cycle 21 startup, the licensee shall submit a request for the staff to review and approve a seismic evaluation to ensure the structural integrity of the main steam line piping from the main steam isolation valves (MSIV) to the main turbine condenser, the main turbine condenser, and the turbine building. The evaluation will be performed to assess the ability of the aforementioned main steam piping and main turbine condenser to remain sufficiently intact to direct main steam leakage from the

MSIVs to the main turbine condenser, consistent with the leakage assumptions in the design-basis accident dose calculations during and after a Safe Shutdown Earthquake. This seismic evaluation will employ an analytical methodology acceptable to the staff and will identify any modifications necessary to support the evaluation. The licensee's approved request shall be fully implemented, including the completion of modifications, within 12 months of approval or prior to CNS Cycle 22 startup, whichever is later.

4.0 SUMMARY

Based on the information provided by NPPD regarding the LOCA and CRDA, the results of the staff's confirmatory calculations, and NPPD's continuing commitment to provide KI to control room personnel, the staff finds reasonable assurance that the postulated radiological consequences of the design-basis LOCA and CRDA at CNS will be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, GDC 19, and Sections 6.4 (LOCA and CRDA) and 15.4.9 (CRDA) of NUREG-0800. Therefore, the changes to the LOCA and CRDA calculational methodologies, as described above, are acceptable.

5.0 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 comment.

6.0 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. 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 (65 FR 4280). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 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.

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Table 1

CNS Loss-of-Coolant Accident Analysis Parameters Used by Staff

Source Term	
Reactor power (2381 x 1.02 (Uncertainty in power measurements)), MWt	2429
Release into primary containment	Instantaneous
Noble gas in containment (Percent of activity in core)	100
Iodine in containment (Percent of activity in core)	25
Iodine species distribution	
Elemental	0.91
Organic	0.04
Particulate	0.05
Release Data	
SGTS Flow, cfm	
0- 1 hours (Each Train)	1492
1 - 720 hours (Idle Train)	288
1 - 720 hours (Operating Train)	1492
SGTS Filter Efficiency, % (Includes 1% filter bypass)	
Idle Train	
Elemental	89
Organic	29
Particulate	94
Operating Train	
Elemental	94
Organic	94
Particulate	94
Primary Containment	
Primary Containment volume, ft^3	239,100
Suppression Pool Decontamination Factor for Elemental and Particulate Iodine	2
Suppression Pool minimum water volume, ft^3	87,650
Mass of fluid in reactor vessel, lb	437,000
Mass of fluid in primary piping system, lb	89,000
Primary Containment Leakage, % volume/ day	0.635
Secondary Containment	
Mixing	No mixing
ESF Release	
ESF Leak Data (directly to SGTS), cc/min	1000
ESF Flashing Fraction, %	10
ESF Source Term, % of core iodine inventory	50

Table 1–Continued

MSIV Leak Data

MSIV leak rate per MSIV, scfh	11.5
Drywell pressure for MSIV leak rate, psia	65
Number of outboard MSIVs	4
Containment temperature for MSIV leak rate, deg. F	309
Standard Pressure, psia	14.7
Standard Temperature, deg. F	60

Control Room

Unfiltered Inleakage, scfm	
Infiltration	71
Ingress/Egress	10
Time to isolate air intake, min	21
Air intake rate, scfm	
0-21 min: Normal Supply	3235
21 min - 30 days: Emergency Supply	900±10%
Control room intake filter efficiency, all species, %	94
Recirculation flow rate, cfm	0
Breathing rate, offsite, 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 volume, ft^3	64640

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
>24 hours	2.32E-4
Atmospheric Dispersion Factors	Table 3

Table 2

CNS Control Rod Drop Accident Analysis Parameters Used by Staff

Source Term	
Reactor power (2381 x 1.02), MWt	2429
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 volume, ft^3	64640
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 (χ/Q) Values Used by Staff (in units of sec./m^3)

Time Period	Control Room	EAB	LPZ
Stack Releases			
0-0.5 hrs	3.03E-4	1.2E-4	1.4E-4
0.5-2 hrs	1.00E-9	1.6E-5	
0.5-8 hrs			4.0E-5
2-8 hrs	2.65E-9		
8-24 hrs	6.41E-8		1.6E-5
1-4 days	2.00E-8		5.8E-6
4-30 days	1.66E-8		1.7E-6
Turbine Building Releases			
0-2 hrs	9.54E-4	5.2E-4	
0-8 hrs			2.9E-4
2-8 hrs	4.93E-4		
8-24 hrs	2.69E-4		7.3E-5
1-4 days	1.72E-4		2.5E-5
4-30 days	1.43E-4		5.2E-6