October 23, 2001

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

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT REGARDING REVISED RADIOLOGICAL DOSE ASSESSMENT AND TECHNICAL SPECIFICATION CHANGES (TAC NO. MB1419)

Dear Mr. Wilson:

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of approval of revisions to the calculational methodology for assessment of radiological consequences of design-basis accidents (DBAs) and changes to Technical Specifications (TSs), in response to Nebraska Public Power District (NPPD, licensee) application dated February 28, 2001, as supplemented by letters dated September 14, 18 and 27, 2001. The letters dated September 14, 18, and 27, 2001 did not alter the conclusions regarding no significant hazard determination consideration.

As a result of its application acceptance review, the Nuclear Regulatory Commission's (NRC's) staff identified issues that required additional time consuming analyses related to revised DBA assessment methodologies for loss-of-coolant accident (LOCA), control rod drive (CRD) accident, and main steam line break (MSLB) accident. Based on the discussions with the licensee, the NRC staff concluded that those issues could not be resolved in time to meet the licensee's schedule for restart of CNS following refueling outage 20. Therefore, the NRC staff and the licensee agreed to defer the final approval of the revised DBA assessment methodologies for LOCA, CRD accident, and MSLB accident to after the restart of CNS from refueling outage 20. The NRC staff's reviews for LOCA, CRD accident, and MSLB accident were directed to interim approval for one operating cycle after restart from refueling outage 20. The review of the deferred methodologies will continue on a preapplication basis, pending the licensee's submittal of the seismic evaluation of the adequacy of the main steam piping, the main steam condenser, and the turbine building.

The amendment approves the DBA assessment methodology for a fuel handling accident, and extends the previous interim approval by an additional operating cycle (Cycle 21). The assessments of a loss-of-coolant accident, and a control rod drop accident are approved with the provision that NPPD will continue to maintain the ability to monitor the radiological conditions during emergencies and administer potassium iodide to control room operators so as to keep the radiological doses of the accidents within the guidelines of the Commission's regulations under 10 CFR Part 50, Appendix A, General Design Criterion 19. The NRC staff has deferred the review of the licensee's revision to the dose assessment methodology of a main steam line break

(MSLB) accident. The NRC staff has, however, determined that continued operation of CNS with the existing acceptable analysis of a MSLB accident is acceptable.

Additionally, the amendment revises the TSs to cause the control room isolation and control room emergency filter system instrumentation to be initiated by the same signals that initiate secondary containment isolation, rather than the present initiation by the control room air inlet radiation monitor. The NRC staff also approves of the elimination of the control room air inlet radiation monitor functions.

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, Section1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 187 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

(MSLB) accident. The NRC staff has, however, determined that continued operation of CNS with the existing acceptable analysis of a MSLB accident is acceptable.

Additionally, the amendment revises the TSs to cause the control room isolation and control room emergency filter system instrumentation to be initiated by the same signals that initiate secondary containment isolation, rather than the present initiation by the control room air inlet radiation monitor. The NRC staff also approves of the elimination of the control room air inlet radiation monitor functions.

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, Section1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 187 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC
PDIV-1 r/f
RidsNrrDlpmLpdiv (SRichards)
RidsNrrDlpmRGramm
RidsNrrPMMThadani
RidsNrrLAMMcAllister
G. Hill (2)
RidsRgn4MailCenter(LHurley, DGraves, KKennedy)
MBlumberg
MHart
LBrown
RidsOgcRp
RidsNrrDripRtsb (WBeckner)
RidsAcrsAcnwMailCenter
HLi

**See previous concurence
*No substantive change from SE input

ACCESSION NO.: ML012960618

OFFICE	PDIV-1/PM	PDIV-1/LA	SPSB/BC*	EEIB/BC*	EMEB/SC**
NAME	MThadani:lf	MMcAllister			KManoly
DATE	10/22/01	10/19/01			10/18/01

OFFICE	RTSB/SC	OGC*	PDIV-1/SC	
NAME	WBeckner	DCummings	RGramm	
DATE	10/22/2001	10/22/2001	10/23/2001	/ /2001

OFFICIAL RECORD COPY

Cooper Nuclear Station

cc:

Mr. G. R. Horn
Sr. Vice President of Energy Supply
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

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

D. F. Kunsemiller, Risk and
Regulatory Affairs Manager
Nebraska Public Power District
P. O. Box 98
Brownville, NE 68321

Dr. William D. Leech
Manager-Nuclear
MidAmerican Energy
907 Walnut Street
P. O. Box 657
Des Moines, IA 50303-0657

Mr. Ron Stoddard
Lincoln Electric System
1040 O Street
P. O. Box 80869
Lincoln, NE 68501-0869

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
Forbes Field Building 283
Topeka, KS 66620

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
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 September 14, 18, and 27, 2001, 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 revision of 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 21.

   B. The license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 187, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

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: October 23, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-61	3.3-61
3.3-62	3.3-62
3.3-63	3.3-63
B 3.3-185	B 3.3-185
B 3.3-186	B 3.3-186
B 3.3-187	B 3.3-187
B 3.3-188	B 3.3-188
B 3.3-189	B 3.3-189
B 3.3-190	B 3.3-190
B 3.3-191	B 3.3-191
B 3.3-192	B 3.3-192
B 3.3-193	B 3.3-193
B 3.4-32	B 3.4-32
B 3.6-67	B 3.6-67
B 3.6-68	B 3.6-68
B 3.6-72	B 3.6-72
B 3.6-73	B 3.6-73
B 3.6-78	B 3.6-78
B 3.6-79	B 3.6-79
B 3.7-17	B 3.7-17
B 3.7-18	B 3.7-18
B 3.9-19	B 3.9-19

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 187

TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

In Amendment No. 183, the Commission approved certain requested changes in design-basis accident (DBA) dose assessment methodology for Cooper Nuclear Station (CNS). The NRC staff and the Nebraska Public Power District (NPPD, licensee) were unable to expeditiously resolve all questions related to assumptions for assessment of consequences of a fuel handling accident (FHA) and main steam line break (MSLB) accident. To permit the licensee to restart CNS from an outage, the Commission deferred the review and approval of changes to FHA and MSLB accidents dose assessment methodologies, deferred the resolution of differences of opinion regarding meteorological dispersion issues, and deferred the condenser structural analysis issues raised by the NRC staff as outlined in the safety evaluation supporting Amendment No. 183. Pending resolution of the above described issues, the Commission approved Amendment No. 183 for interim operation of CNS for one operating cycle only.

By letter dated February 28, 2001, as supplemented by letters dated September 14, 18, and 27, 2001, the licensee resubmitted its request for amendment to License DRP-46 to revise the CNS calculation methodology for assessment of radiological consequences of DBAs. The licensee requested approval of the revised assumptions and revised postulated source term for assessment of consequences of FHA, loss-of-coolant accident (LOCA), control rod drop (CRD) accident, and MSLB accident. The licensee also submitted responses to meteorological dispersion issues raised during the NRC staff's evaluation for Amendment No. 183. The response to issues regarding condenser structural adequacy was not complete and was deferred for later NRC staff evaluation.

Additionally, the licensee requested changes to the Technical Specifications (TSs) to address planned modifications to install the control room emergency filter (CREF) system instrumentation initiation. The licensee plans to eliminate the radiation monitor in the control room inlet air as initiator of control room isolation and to pressurize the control room. Instead, the licensee proposes to isolate and pressurize the control room based on the signals that initiate the secondary containment isolation. Accordingly, the licensee has requested changes to the CNS TSs to require the isolation of control room and initiation of the CREF system by the same signals that initiate secondary containment isolation.

As a result of its application acceptance review, the Commission's staff identified issues that required additional time consuming analyses related to revised DBA assessment methodologies for LOCA, CRD accident, and MSLB accident. Based on the discussions with the licensee, the NRC staff concluded that those issues could not be resolved in time to meet the licensee's schedule for restart of CNS following refueling outage 20. Therefore, the NRC staff and the licensee agreed to defer the final approval of the revised DBA assessment methodologies for LOCA, CRD accident, and MSLB accident to after the restart of CNS from refueling outage 20. The NRC staff's reviews for LOCA, CRD accident, and MSLB accident were directed to interim approval for one operating cycle after restart from refueling outage 20. The review of the deferred methodologies will continue on a preapplication basis, pending the licensee's submittal of the seismic evaluation of adequacy of the main steam piping, the main steam condenser, and the turbine building.

2.0 EVALUATION

The NRC staff has reviewed the licensee's application for approval of the revised calculational methodology for assessment of radiological consequences due to postulated FHA, CRD accident, MSLB accident, and LOCA. The licensee's meteorological dispersion calculations are also being reviewed. The NRC staff performed independent evaluation of postulated DBAs to determine the licensee's continued compliance with the Commission's regulations under 10 CFR Part 100; and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19. The design inputs utilized by the NRC staff to evaluate these accidents are given in Tables 1 through 4. Inputs and model development issues that warrant further discussion are discussed below.

The control room inlet air radiation monitor signal causes the control room isolation and control room emergency filter system initiation. However, the licensee has now proposed TS changes, by which signals that initiate secondary containment isolation also will trigger the control room air inlet radiation monitor initiation functions. The staff has evaluated this change.

2.1 Revised Calculation Methodology for Assessment of Radiological Consequences of DBAs

NPPD evaluated the impact of the proposed changes in methodology to ensure 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 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 analysis methodology.

2.1.1 Fuel Handling Accident Radiological Analysis

The licensee made several changes to the radiological analysis of FHA. In the course of its review, the NRC staff requested additional information from the licensee regarding its assumptions of: (1) a time variable release from the reactor building vent during fan coastdown prior to secondary containment isolation, (2) the control room isolation and CREF system initiation in 10 seconds, and (3) control room atmospheric dispersion factors for the reactor building vent release. For discussion of the NRC staff’s review of atmospheric dispersion factors, see Section 2.1.3, “Meteorology Considerations,” below. The licensee based the

variable release rate from the reactor building on the vendor-supplied exhaust fan coastdown curve and exhaust fan characteristics, standard fan laws, and exhaust ductwork flow resistance characteristics. Modification to the CREF system initiation logic is the basis for the licensee's assumption that the CREF System initiates in 10 seconds. Testing and verification of this initiation time will be done after installation and implementation of changes to TS 3.3.7.1. The licensee stated that the CREF System initiation signal is from the same radiation monitor already used for the secondary containment isolation signal, "Reactor Building Exhaust Plenum Radiation - High," which is included in TS 3.3.6.2.

With regard to the proposed TS 3.3.7.1 allowed value of $\leq$ 49 mR/hr for the "Reactor Building Ventilation Exhaust Plenum Radiation - High" function, the NRC staff questioned whether this allowed value for the radiation monitor would detect the postulated release from the FHA and enable the CREF System to be initiated as assumed in the licensee's FHA radiological analysis. The licensee stated that the TS 3.3.7.1 allowable value was derived using the General Electric (GE) setpoint methodology based on the analytical values developed by GE for CNS, and was chosen to promptly detect gross failure of the fuel cladding. The licensee used the same allowable value for secondary containment isolation that is in existing TS 3.3.6.2, since the same instrumentation will be used for both purposes. The revised FHA analysis assumes that a higher number of curies of iodine and noble gases are released to the secondary containment during the first minute of the postulated accident than the current FHA of record in the CNS updated safety analysis report. Therefore, the NRC staff agrees that the allowable value of $\leq$ 49 mR/hr remains conservative relative to revised FHA.

The NRC staff performed confirmatory calculations using licensee's assumptions, which are listed in Table 3. These calculations confirmed the licensee's dose results. The offsite dose consequences of the licensee's revised FHA meet the NUREG-0800 Standard Review Plan 15.7.4 acceptance criteria and are well within the dose limits given in 10 CFR Part 100. The control room dose consequences calculated by the licensee are within the dose guidelines given in GDC-19. For the reasons stated above, the NRC staff found the licensee's analysis of FHA acceptable.

2.1.2 Main Steam Line Break (MSLB) Radiological Analysis Review Deferred

The NRC staff has deferred review of the licensee's MSLB accident analysis, submitted on February 28, 2001, until a later date. Although the NRC staff has not fully reviewed the MSLB analysis submittal, the NRC staff has determined deferral was acceptable because the proposed changes to the CREF system initiation logic would not impact the analysis of the radiological consequences of the MSLB. Changes to the CREF initiation logic would not affect assumptions for the offsite dose analysis, and the source term for the MSLB has not been significantly changed from that documented in the CNS final safety analysis report (FSAR). Therefore, the NRC staff finds that the existing MSLB analysis in the CNS FSAR remains bounding. The licensee's current FSAR does not include a control room habitability analysis of the MSLB. The licensee's calculation of the control room dose for the MSLB, submitted February 28, 2001, does not take credit for CREF initiation. As discussed below in Section 2.1.5, "Control Room Habitability Generic Issue," the licensee continues to have compensatory measures available to mitigate the control room radiological consequences in the event of a radiological release.

Though the NRC staff has not fully reviewed the revised MSLB analysis at this time, because of the above discussion, the NRC staff has reasonable assurance that the radiological consequences of the MSLB remain acceptable.

2.1.3 Meteorology Considerations

2.1.3.1 Meteorological Data

The licensee used hourly onsite meteorological data collected during calendar years 1994 through 1998 other than for fumigation conditions assumed to occur from the stack, to calculate the control room relative concentration (X/Q) values, for LOCA, CRD accident, and FHA. Meteorological measurements used in these calculations were made at the 10, 60, and 100 meter levels. For the 5 year period, joint wind speed, wind direction, and atmospheric stability data recovery was less than the recommended minimum of 90 percent cited in Regulatory Guide 1.23, “Onsite Meteorological Programs.” The NRC staff review of the data indicated that this was primarily due to lower data recovery in 1995 and 1996 with respect to temperature difference (ΔT) measurements and, to a lesser extent, some of the wind direction measurements. However, data recovery for wind speed was well above the 90 percent minimum each year. The licensee noted that instrument accuracy limits for ΔT 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 would not affect determination of the atmospheric stability category. Aside from these exceptions, the licensee confirmed that the meteorological measurement program met the guidance provided in Regulatory Guide 1.23.

The NRC staff performed a review of the data and found good consistency of wind speed and wind direction data between both the two measurement heights and from year to year. Temperature difference data did not appear as consistent. There was a higher occurrence of neutral (Class D) conditions in 1996 and there was considerably more extremely unstable (Class A) conditions in 1997 and 1998 than in the other years. The NRC staff also noted occurrence of extremely unstable conditions at night. The licensee attributed these variations to factors such as climatological variability, wind shifts, and minor temperature fluctuations. These variations should not have a significant impact on the calculation of the 95 percentile X/Q values used in the accident assessments described above.

2.1.3.2 Relative Concentration Estimates for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ)

For LOCA, CRD accident, and FHA dose assessment, the licensee calculated X/Q values for the EAB and 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 reactor building vent and 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.

2.1.3.3 Relative Concentration Estimates for the Control Room

The licensee 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 LOCA, CRD accident, and FHA control room dose assessments, other than for fumigation conditions. The modifications resulted from discussions with the NRC staff regarding fumigation, initial diffusion coefficients, and the vent release option. Initially, the licensee made X/Q estimates for each of the 5 years individually using the ARCON96 computer code and selected the highest values irrespective of year. When a data processing error was subsequently discovered, the licensee recalculated X/Q values using the entire 5 year interval, compared the results with the previously calculated year-by-year estimates, and used the higher values in its dose assessment.

For postulated releases from the reactor building vent to the control room, the licensee assumed that no effluent would be released to the environment during the first 3.8-seconds of the accident. The licensee postulated that the reactor building vent exhaust fans running prior to the accident would lose power and coast down. For the next 86.2 seconds, effluent would be released from the reactor building vent as a ground level release. The licensee estimated the effect of the fan coast down by making calculations at two different fan flow rates and applying each of the resulting two X/Q values to part of the period. After the initial 90 second period, effluent was postulated to be released from the plant stack as an elevated release, with fumigation occurring during the first 30-minutes of the accident. The fumigation value used in the dose assessment is that provided previously by the licensee in the CNS post TMI-requirements/action plan.[1]

Using the ARCON96 methodology, the licensee 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 plenum located further away from the control room than the turbine building. 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 the GDC 19. Applicability of the diffuse release assumption to the Cooper turbine building is still under review. Diffuse source modeling should only be used for those situations in which the radioactivity being released is homogeneously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building. Since leakage is more likely to occur at a penetration, the potential impact of building penetrations exposed to the environment must be considered. If the penetration release would be more limiting, the diffuse area source assumption should not be used. The NRC staff's estimates assuming a point release from the side of the turbine building nearest the control room intake

---

[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.

indicate that the resultant dose, while higher than calculated assuming a diffuse source, is still within guidelines of GDC 19.

2.1.3.4 Meteorology Conclusions

The NRC staff finds the ground level and stack continuous release EAB and LPZ X/Q values calculated by the licensee for CNS acceptable for use in the dose assessment described herein. The NRC staff also finds the control room X/Q values from the reactor building vent and plant stack acceptable. The NRC staff finds the turbine building diffuse release X/Q values acceptable on an interim basis until refueling outage 21.

2.1.4 Compensatory Actions for the LOCA analysis

The CNS evaluations did not apply any credit for mitigation of control room dose by utilizing potassium iodine, although they committed to continuing an interim compensatory measure to provide reasonable assurance that GC 19 guidelines are met. The interim compensatory measure is to continue implementation of a previous commitment to make potassium iodide tablets available to the control room operators if plant conditions indicate that a LOCA is occurring coincident with core damage.

Using the LOCA analysis parameters provided by CNS (Table 1), the NRC staff's evaluation of the structural integrity of main steam piping and main turbine condenser following a Safe Shutdown Earthquake[2]; and applying a protection factor of 10 for utilization of potassium iodide; the NRC staff performed its own evaluation of the control room operator doses. Based upon comparison of the CNS evaluation and the NRC staff's results, the NRC staff agrees, that by utilizing potassium iodide, CNS can meet GC 19 requirements. The NRC staff considers the use of potassium iodide acceptable until CNS enters mode 4 in preparation for refueling outage 21 (approximately one operating cycle). This will facilitate the resolution of issues concerning the CNS ARCON96 turbine building releases, and full qualification of the seismic adequacy of the main steam piping, main turbine condenser, and the turbine building can be resolved.

2.1.5 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. Twenty-two control rooms have been tested using enhanced test methods. In all 22 cases, the measured infiltration rates exceeded the values assumed in the design-basis analyses. In each case the affected licensee was able to either reduce the excessive infiltration or show the acceptability of the observed infiltration. However, the collective experience has caused concerns regarding those facilities that have not performed the enhanced testing. The NRC staff is currently working to resolve these concerns.

---

[2] "Cooper Nuclear Station - Issuance of Amendment on Design Basis Accident Radiological Assessment Calculational Methodology Revision, TAC No. MA7758," dated April 7, 2000.

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 NRC staff bases this determination on the availability of potassium iodide 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.

2.1.6 Conclusions Regarding Evaluation of Revised DBA Methodology

Based on the considerations discussed above, the information provided by NPPD regarding LOCA, CRD accident and FHA, and NPPD's continuing commitment to provide potassium iodide to control room personnel, the NRC staff, finds reasonable assurance that the postulated radiological consequences of the design basis LOCA, CRD accident and FHA will be less than the dose criteria 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 consideration discussed above, that the postulated radiological consequences of the design basis LOCA, CRD accident and FHA are within the guidelines of 10 CFR Part 100 and Sections 15.4.9 (CRD accident) and 15.7.4 (FHA) of NUREG-0800. Accordingly, the NRC staff approves FHA, extends the approved of LOCA, and CRD accident methodologies for one operating cycle and defers the review of MSLB accident methodology.

2.2 Technical Specification Changes-Control Room Emergency Filtration System

The licensee also proposed changes to TS 3.3.7.1, CREF system instrumentation. The existing TS 3.3.7.1 requires isolation and pressurization of the control room based on a signal from the control room air inlet radiation monitors. The proposed TS will require isolation and pressurization of the control room based on the same signals that initiate secondary containment isolation. These signals include the Reactor Vessel Water Level-Low (Level 3), the Drywell Pressure-High, and the Reactor Building Ventilation Exhaust Plenum Radiation-High. The existing control room air inlet radiation monitoring function will be eliminated.

The Commission's regulatory requirements related to the content of the TS are set forth in regulation 10 CFR 50.36, "Technical Specifications." This regulation requires that the TS include items in five categories. These categories are: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the regulation does not specify the particular TS to be included in a plant's license.

The current CNS DBA radiological assessment of control room personnel exposure has been based on either sensing high radiation in the normal control room air inlet supply to initiate isolation of the control room environment from the normal air intake, and to initiate the CREF system to pressurize the control room and filter incoming air, or on manual initiation of the CREF System. The revised DBA radiological assessment calculation no longer credits the control room air intake radiation monitors for initiation of control room isolation and CREF System actuation. The revised DBA radiological assessment methodology will use the Reactor Vessel Water Level-Low (Level 3), the Drywell Pressure-High, and the Reactor Building Ventilation Exhaust Plenum Radiation-High signals. The first two signals are indicative of a

LOCA while the third signal is indicative of an imminent release of radiation to the environment, such as the release following a fuel handling accident. Since these signals are the same signals utilized for the Secondary Containment Isolation Instrumentation (TS 3.3.6.2) and the signal for these functions will be from the same sensors, therefore; the proposed LCO, applicability, actions, and SRs for TS 3.3.7.1 will be similar to TS 3.3.6.2.

The licensee plans to implement this design change following the approval of the TS change. This design will consist of two trip systems. Each trip system includes the sensors, relays, and switches necessary to actuate the CREF System. Each trip system receives input signals from four sensors conforming with the parameters listed above. Each sensor sends a signal to both trip systems.

The Reactor Vessel Water Level-Low (Level 3), Drywell Pressure High, and Reactor Building Ventilation Exhaust Plenum Radiation-High functions are each arranged in a one-out-of-two taken twice logic for each trip system.

The proposed TS 3.3.7.1 changes are generally consistent with NUREG-1433, BWR/4 Standard Technical Specifications. A few differences are due to plant-specific design configuration differences with the NUREG-1433 template. The licensee has addressed and justified these differences. The NRC staff finds the justifications are acceptable.

The CREF is a single train system, and the existing initiation instrumentation is included in a single one-out-of-three trip system. The current Actions for inoperable instrumentation channels are simply to manually actuate the CREF System or to declare the CREF System inoperable. With the proposed initiation functions, diversity and redundancy of sensors are available for CREF System initiation. The proposed required actions for inoperable CREF System instrumentation channels provide appropriate measures for separate inoperable channels. The actions are modified by a note allowing separate condition entry for each inoperable channel. The actions have also been revised to allow restoration or placing channel(s) in trip within completion time of 12 hours for Functions 1 and 2, and 24 hours for Function 3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure in the trip system, and allow operation to continue. Alternately, if it is not desired to place the channel in trip, Condition C must be entered and its required actions taken. The 12 and 24 hour completion times are consistent with the times allowed by TS 3.3.6.2 for the same channels. These allowable times were approved by NRC in July 1992 for GE Topical Report GENE-770-06-01, “Basis for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications.”

The setpoint methodology properly accounts for instrument drift and the associated channel functional test and channel calibration frequencies of 92 days and 18 months, respectively. The setpoint calculations are performed using methodology described in NEDC-31336P-A, “General Electric Instrument Setpoint Methodology,” dated September 1996.

The proposed Bases for TS 3.3.7.1 reflect the changes to the dose calculations and the corresponding changes to TS 3.3.7.1. Formal approval of these changes will be made in accordance with the provision of TS 5.5.10, Technical Specification Bases Control Program.

Paragraph (c)(2)(ii) of 10 CFR 50.36 provides screening criteria for items to be removed from the plant technical specifications. By letter dated September 14, 2001, the licensee addressed the removal of the control room air inlet radiation-high function and replaced it with signals that initiate secondary containment isolation function. The control room air inlet radiation-high signal is no longer credited for protection of the control room personnel. Control room personnel protection is provided by actuation of the control room emergency filter system on reactor vessel water level-low (Level 3), drywell pressure-high, and reactor building ventilation exhaust radiation-high. The licensee has compared the control room air inlet radiation-high function with four criteria listed in paragraph (c)(2)(ii) of 10 CFR 50.36 and concluded that since the screening criteria have not been satisfied, the control room air inlet radiation-high function may be removed from the CNS TS. The NRC staff finds this acceptable.

The NRC staff has reviewed the proposed changes on TS 3.3.7.1, "Control room emergency filter system instrumentation." Three functions to isolate and pressurize the control room are identical to the functions that isolate the secondary containment. The proposed LCO, applicability, actions, and SRs for TS 3.3.7.1 are similar to TS 3.3.6.2, Secondary containment isolation instrumentation." The licensee has documented the justification for removal of the control room air inlet radiation-high function from TS 3.3.7.1 in according with Paragraph (c)(2)(ii) of 10 CFR 50.36 screening criteria. The NRC staff concludes that the proposed changes on TS 3.3.7.1 are acceptable.

3.0 STATE CONSULTATION

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

4.0 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 (66 FR 48288). 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.

5.0 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.

Attachments:
1. Table 1
2. Table 2
3. Table 3
4. Table 4

Principal Contributors: M. Blumberg
M. Hart
L. Brown
H. Li

Date: October 23, 2001

Table 1
CNS Loss of Coolant Accident Analysis Parameters Used by NRC 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)	
<u>Idle train</u>	
Elemental	89
Organic	29
Particulate	94
<u>Operating train</u>	
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

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, sec	11
Air intake rate, scfm	
0-11 sec: Normal supply	3235
11 sec - 30 days: emergency supply	900±10%
Control room intake filter efficiency, all species, percent	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 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
>24 hours	2.32E-4
Atmospheric dispersion factors	Table 4

Table 2
CNS Control Rod Drop Accident Analysis Parameters Used by NRC 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 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 4

Table 3
CNS Fuel Handling Accident Analysis Parameters Used by NRC Staff

Source Term

Reactor power (2381 x 1.02), MWt	2429
Decay time after shutdown, hr	67
Radial peaking factor	1.8
Number of failed fuel rods	151
Number of fuel rods per bundle	87.333
Number of fuel bundles in core	548
ORIGEN2 core radionuclide inventory	
Fuel rod gap activity release fractions, %	
Noble gases except Kr-85	10
Kr-85	30
Iodines except I-131	10
I-131	12
Pool iodine decontamination factor	100
Iodine form released from pool, %	
Elemental	75
Organic	25

Release Data

Isolation of secondary containment, sec	90
Time variable exhaust fan flow from reactor building vent	
During coastdown to isolation (0-90 sec)	
(See licensee's calculation NEDC 99-032, Rev 2)	
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)	
<u>Idle train</u>	
Elemental	89
Organic	29
Particulate	94
<u>Operating train</u>	
Elemental	94
Organic	94
Particulate	94

Secondary Containment

Mixing	No mixing

Control Room

Unfiltered inleakage, scfm	
Infiltration	71
Ingress/Egress	10
Time to isolate air intake, sec	10
Air intake rate, scfm	
0 -10 sec: Normal supply	3235
10 sec - 30 days: Emergency supply	900±10%
Control room intake filter efficiency, all species, percent	94
Recirculation flow rate, cfm	0
Breathing rate, control room, 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
>24 hours	2.32E-4
Atmospheric dispersion factors	Table 4

Table 4

CNS Atmospheric Relative Concentration (X/Q) Values Used by NRC Staff

The NRC staff finds use of the following EAB, and LPZ X/Q values (sec/m^3) acceptable for postulated ground level releases from the reactor and turbine buildings and the elevated plant stack.

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 reactor building vent and elevated plant stack releases.

Receptor Location	Ground level 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
2 - 8 hrs		2.65 E-9
8 - 24 hrs		6.41 E-8
1 - 4 days		2.00 E-8
4 - 30 days		1.66 E-8

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