Nebraska Public Power District
Nebraska's Energy Leader

NLS990122
December 22, 1999

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Design Basis Accident Radiological Assessment Calculational Methodology Revision
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Reference: 1. Nebraska Public Power District Letter (NLS990067) from John H. Swailes to U.S. Nuclear Regulatory Commission, dated August 2, 1999, "Withdrawal of Proposed License Amendment, Secondary Containment Isolation Description and Refueling Accident Analysis Results."

In accordance with the provisions specified in 10 CFR 50.4 and 10 CFR 50.90, the Nebraska Public Power District (District) submits a request for an amendment to License DPR-46 to revise the Cooper Nuclear Station (CNS) design basis accident radiological assessment calculational methodology used to demonstrate compliance with the Exclusion Area Boundary and Low Population Zone dose acceptance criteria specified in 10 CFR 100.11, and the control room dose acceptance criteria discussed in General Design Criterion 19 (GDC 19) of 10 CFR 50, Appendix A.

In Reference 1 the District indicated, in part, that due to inconsistencies identified between the refueling accident control room operator dose and off-site dose calculations, and differences between the District and Nuclear Regulatory Commission (NRC) staff regarding the assumptions used for calculating doses to the control room operators following a postulated loss of coolant accident, a proposed license amendment revising the CNS Design Basis Accident (DBA) radiological assessment calculations would be submitted to the NRC. It was also indicated in Reference 1 that the proposed amendment would address the recently issued Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal" (GL 99-02), and an Unreviewed Safety Question (USQ) associated with the CNS Secondary Containment isolation description and Refueling Accident analysis results. Subsequent to the issuance of Reference 1, the District determined that inclusion of higher burnup fuel design considerations (i.e., GE14 fuel design) into the CNS DBA radiological assessment should be conducted concurrent with the revised

Cooper Nuclear Station
P.O. Box 98 / Brownville, NE 68321-0098
Telephone: *(402) 825-3811 /* ***Fax:*** *(402) 825-5211*
http://www.nppd.com

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DBA radiological assessment calculational methodology submittal to support startup from the CNS spring 2000 refueling outage and ensure that fuel management can be optimized in subsequent fuel operating cycles.

Discussions were held between the District and the NRC CNS Project Manager during November 1999 regarding the proposed submittal discussed in Reference 1, and incorporation of higher burnup fuel design considerations into the revised CNS DBA radiological assessment calculational methodology. Per discussions with the NRC, the District is submitting the revised CNS DBA radiological assessment calculational methodology as a stand-alone submittal to facilitate review of the methodology and support a startup during the week beginning April 2, 2000. The remaining issues identified in Reference 1 will be addressed via separate submittals as discussed with the NRC CNS Project Manager during November and December 1999. The District requests issuance of the proposed license amendment by March 31, 2000, to support the CNS Cycle 20 startup, currently scheduled for April 3, 2000.

The CNS DBA radiological assessment offsite and control room operator dose calculations have been revised in their entirety and are contained in Attachment 2. The reasons for revising the accident dose calculations in their entirety are to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of CNS's vintage, incorporate the Technical Information Document (TID-14844) source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. Due to the broad changes in the calculational and software methodology, DBA initial assumptions, and an increase in the postulated accident source term, the calculated radiological dose consequences of each design basis accident have changed, and in certain cases increased. In each case, however, the calculated radiological dose assessment consequences satisfy 10 CFR 100.11 and GDC 19 dose acceptance criteria. Because of the changes in methodology and resulting dose assessment consequences, these revisions are considered to represent an unreviewed safety question under 10 CFR 50.59 and are being submitted to the NRC for review and approval under 10 CFR 50.90.

Attachment 1 contains a general description of the proposed revision to the CNS DBA radiological assessment calculational methodology as well as the supporting 10 CFR 50.92 and environmental impact evaluations. Attachment 2 contains the proposed revisions to the CNS DBA radiological assessment calculational methodology. The proposed license amendment revision has been reviewed by the necessary safety review committees and incorporates all amendments to the CNS Facility Operating License through Amendment 179 issued July 26, 1999. The District has concluded that the proposed changes do not involve a significant hazards consideration.

By copy of this letter and attachments the appropriate State of Nebraska official is being notified in accordance with 10 CFR 50.91(b)(1). Copies to the Region IV Office and the CNS Resident Inspector are also being sent in accordance with 10 CFR 50.4(b)(2).

Should you have any questions concerning this matter, please contact Mr. Paul Caudill at (402) 825-5052.

Sincerely,

John H. Swailes
Vice President of Nuclear Energy

/rlb

Attachments

cc: Regional Administrator w/attachments
USNRC - Region IV

Senior Project Manager w/attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachments
USNRC

Environmental Health Division- Program Manager w/attachments

NPG Distribution w/o attachments and enclosures

STATE OF NEBRASKA)
)
NEMAHA COUNTY)

John H. Swailes, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this correspondence on behalf of Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

John H. Swailes

Subscribed in my presence and sworn to before me this 22nd day of December, 1999.

Wilma M Werner
NOTARY PUBLIC

Wilma M. Werner
General Notary
State of Nebraska
My Commission Expires Oct. 26, 2002

ATTACHMENT 1
PROPOSED LICENSE AMENDMENT TO REVISE THE DESIGN BASIS RADIOLOGICAL ASSESSMENT CALCULATIONAL METHODOLOGY
COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

1.0 INTRODUCTION

The Nebraska Public Power District (District) requests that the Nuclear Regulatory Commission (NRC) review and approve the revised Cooper Nuclear Station (CNS) design basis accident radiological assessment calculational methodology contained in Attachment 2. Because of the broad changes in methodology these revisions are considered to represent an unreviewed safety question under 10 CFR 50.59 and are being submitted to the NRC for review and approval under 10 CFR 50.90. The revisions to the radiological assessment calculational methodology resulted in changes to the calculated radiological assessment consequences for each design basis accident. In some cases the consequences have increased, but in each case the calculated radiological dose consequences satisfy the Exclusion Area Boundary and Low Population Zone radiological dose acceptance criteria specified in 10 CFR 100 and the control room dose acceptance criteria discussed in General Design Criteria 19 (GDC 19) of 10 CFR 50, Appendix A. Following NRC approval of this change, the CNS Updated Safety Analysis Report (USAR) will be changed to reflect the revised design basis accident radiological assessment.

2.0 DISCUSSION

Due to inconsistencies identified between the refueling accident control room operator dose and off-site dose calculations, and differences between the District and NRC staff regarding the assumptions used for calculating doses to the control room operators following a postulated loss of coolant accident, a proposed license amendment revising the CNS Design Basis Accident (DBA) radiological assessment calculations was planned for submittal to the NRC. It was intended that the proposed license amendment would also address the recently issued Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal" (GL 99-02), and an Unreviewed Safety Question (USQ) associated with the CNS Secondary Containment isolation description and Refueling Accident analysis results. In August 1999, the District determined that inclusion of higher burnup fuel design considerations into the CNS DBA radiological assessment should be conducted concurrent with the revised DBA radiological assessment calculational methodology submittal, to support startup from the CNS spring 2000 refueling outage and ensure that fuel management can be optimized in subsequent fuel operating cycles.

Discussions were held between the District and the NRC CNS Project Manager during November 1999, regarding the proposed submittal discussed in Reference 8.6, and incorporation of higher burnup fuel design considerations into the revised CNS DBA radiological assessment calculational methodology. Per discussions with the NRC, the District is submitting the revised CNS DBA radiological assessment calculational methodology as a stand-alone submittal to facilitate review of the methodology and support a startup during the week beginning April 2, 2000.

The remaining issues identified in Reference 8.6 will be addressed via separate submittals as discussed with the NRC CNS Project Manager during November and December 1999.

The reasons for revising the accident dose calculations in their entirety, rather than only correcting the calculational inconsistencies, are to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of CNS's vintage, incorporate the Technical Information Document (TID-14844) source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs such as the GE14 fuel, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. Due to the broad changes in the calculational methodology and an increase in the postulated accident source term, the calculated radiological dose consequences of each DBA have changed and in some cases increased. In each case, however, the calculated radiological dose consequences satisfy the Exclusion Area Boundary and Low Population Zone radiological dose acceptance criteria specified in 10 CFR 100 and the control room dose acceptance criteria discussed in GDC 19 of 10 CFR 50, Appendix A.

3.0 DESCRIPTION OF CHANGES

The proposed license amendment requests NRC review and approval of revisions to the CNS DBA radiological assessment calculational methodology used to demonstrate compliance with the Exclusion Area Boundary and Low Population Zone dose acceptance criteria specified in 10 CFR 100.11, and the control room dose acceptance criteria discussed in GDC 19 of 10 CFR 50, Appendix A. The revisions, as described previously, entail a complete rewrite of the radiological assessment calculational methodology. The proposed changes do not revise the accident category, general accident description, identification of accident cause, frequency classification, starting conditions of the accident, accident sequence of events and system operation as described in the CNS USAR. The revised radiological assessment calculational methodology does, however, involve changes to the radiological consequence summary, fission product release from fuel assumptions, fission product release to secondary containment assumptions and conditions, fission product release to the environs assumptions and initial conditions, and radiological effects summary described in the CNS USAR. Additionally, the revised CNS DBA radiological assessment calculational methodology incorporates the GDC 19 control room dose acceptance criteria determination as part of the assessment. Previously the control room dose assessment was maintained as separate design calculations and not included in the CNS USAR DBA radiological assessment summaries.

The revised CNS design basis accident radiological assessment calculational methodology is included in Attachment 2 for your review. Associated USAR changes will be processed in accordance with 10 CFR 50.71(e) pending approval of this proposed license amendment.

4.0 JUSTIFICATION

Due to the vintage of CNS, many of the existing CNS USAR DBA radiological assessment assumptions and conditions are based on boiling water reactor radiological consequence assumptions and analysis from the late 1960's and early 1970's. In the years following CNS initial licensing additional calculations were performed to assess control room operator dose consequences following a DBA. These calculations used revised radiological assessment assumptions based on updated plant design, licensing requirements applicable to the control room radiological analysis, and CNS specific license amendments.

Radiological dose consequence reviews, conducted in support of CNS license change requests and facility modification evaluations, are extremely time consuming and difficult to conduct as a result of the uniqueness of the CNS DBA radiological assessment calculations, methodologies, and associated format. In mid-1999 the District determined that a complete revision of the CNS DBA radiological assessment calculational methodology was necessary to improve the degree of rigor, completeness, consistency, and useability of the assessment methodology.

The CNS USAR DBA radiological assessment calculational methodology has, therefore, been revised in its entirety. Radiological consequence assumptions and initial conditions were standardized when possible. Assumptions, initial conditions, and calculational methodologies were replaced, in many cases, with readily available and accepted information contained in Regulatory Guides, NUREGs, and TID-14844. Applicable CNS specific plant system design and licensing basis information were also used to replace generic boiling water reactor assumptions contained in the original DBA radiological assessment calculations. Updated meteorological data obtained from recent CNS Radioactive Effluent Release Reports and evaluated per NUREG/CR-6331 (ARCON96) replaces historical meteorological data obtained from the CNS USAR. A newer accident analysis code (AXIDENT) which has been used by other licensees in recent license change requests to the NRC (Crystal River, Brunswick, Quad Cities) replaces the methodology used in the original CNS licensing DBA analysis. Higher burnup fuel considerations were also incorporated into the radiological assessment calculational methodology.

Additionally, the revised CNS DBA radiological assessment calculational methodology incorporates the GDC 19 control room dose acceptance criterion determination. Previously the control room dose assessment was maintained as separate design calculations and not included in the CNS USAR DBA radiological assessment summaries. As in the case of the revisions to the radiological assessment calculational methodology discussed above, the goal for updating and incorporating the control room dose assessment into the overall CNS DBA radiological assessment summary was to improve the degree of rigor, completeness, consistency, and useability of the control room dose assessment. Inclusion of the control room dose assessment calculations in the CNS USAR DBA radiological assessment ensures consistency in assumptions, initial conditions, plant response assumptions and calculational methodology.

The revised CNS DBA radiological assessment calculational methodology will also support upcoming license change requests involving a Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal" (GL 99-02) required Technical Specification change, resolution of an Unreviewed Safety Question (USQ) associated with the CNS secondary containment isolation description and associated refueling accident analysis results, and an interim compensatory action currently in place at CNS resulting from the NRC Safety Evaluation Report for Technical Specification Amendment 167.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazards posed by the issuance of the amendment. This evaluation is to be performed with respect to the criteria given in 10 CFR 50.92(c). The following analysis meets those requirements.

Evaluation of this Amendment with Respect to 10 CFR 50.92

1. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revisions to the Design Basis Accident (DBA) radiological assessment calculational methodology do not affect the accident initiators or precursors of accidents previously evaluated. The proposed revisions to the methodology do not affect the existing design, function or operation of systems, structures or components in the facility. No new or different type of plant equipment is installed by the revised radiological assessment calculational methodology. Plant operating modes are not changed due to the proposed revision to the DBA radiological assessment calculational methodology. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the Technical Information Document (TID-14844) source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. The revisions utilize conservatively lower accident mitigation system filter efficiency assumptions and incorporate plant specific accident mitigation system operating parameter and design assumptions which result in a calculated radiological consequence increase. Operation of accident mitigation systems, structures and components is not altered by the changes in accident mitigation assumptions. Due to the broad changes in the calculational methodology and assumptions, and an increase in the postulated accident source term, the calculated radiological dose consequences of each design basis accident have changed and in some cases increased. In each case, however, the calculated radiological dose consequences satisfy the Exclusion Area Boundary and Low Population Zone radiological dose acceptance criteria specified in 10 CFR 100 and the control room dose acceptance criteria discussed in General Design Criteria 19 (GDC 19) of 10 CFR 50, Appendix A. Therefore, the proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does not create the possibility for a new or different kind of accident from any accident previously evaluated.

The proposed revisions to the DBA radiological assessment calculational methodology do not change the existing design, function or operation of systems, structures or components in the facility. No new or different type of plant equipment is installed by this change. There are no changes to existing design parameters governing plant operation, plant operating modes, or changes in system interfaces. No new types of accident initiators or precursors are created by the proposed revision to the DBA radiological assessment calculational methodology. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the TID-14844 source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. Therefore, the proposed change does not create the possibility of a new or different kind of accident previously evaluated.

3. Does not create a significant reduction in the margin of safety.

The proposed revisions to the DBA radiological assessment calculational methodology do not involve a relaxation in the criteria used to establish safety limits or a relaxation in the limiting conditions for operation. The accident analysis sequence of events remains unchanged. The proposed change will not result in any challenges to plant equipment, fuel integrity, or the reactor coolant system pressure boundary. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the TID-14844 source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

6.0 ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amount of any effluents that may be released off-site, or (3) result in an increase in individual or cumulative occupational radiation exposure. The District has reviewed the proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in

10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the proposed license changes. The basis for this determination is as follows:

1. The proposed license amendment does not involve significant hazards as described previously in the No Significant Hazards Consideration Evaluation.

2. As discussed in the No Significant Hazards Consideration Evaluation, this proposed change does not result in a significant increase in radiological doses for any Design Basis Accident. This proposed change does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site. The proposed license amendment does not introduce any new equipment, nor does it require any existing equipment or systems to perform a different type of function than they are presently designed to perform. The District has concluded that there will not be a significant increase in the types or amounts of any effluents that may be released off-site and these changes do not involve irreversible environmental consequences beyond those already associated with normal operation.

3. The proposed change involves a revision to the Cooper Nuclear Station design basis accident radiological assessment calculational methodology. As discussed in the No Significant Hazards Consideration Evaluation, this change does not affect plant systems or operation. Thus, the proposed change does not increase individual or cumulative occupational radiation exposure beyond that already associated with normal operation.

7.0 CONCLUSION

The District has evaluated the proposed changes to the Cooper Nuclear Station design basis accident radiological assessment calculational methodology against the criteria given in 10 CFR 50.92 (c) in accordance with the requirements of 10 CFR 50.91(a)(1). This evaluation has determined that the proposed changes will not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility for a new or different kind of accident from any previously evaluated; or 3) create a significant reduction in the margin of safety. Therefore, for the reasons detailed above, the District requests NRC approval of this proposed amendment.

8.0 REFERENCES

8.1 10 CFR Part 50, Sections 50.4, 50.59, 50.71(e), 50.90, 50.91, 50.92, and Appendix A

8.2 10 CFR 51.22

8.3 10 CFR 100.11

8.4 CNS Updated Safety Analysis Report (USAR) Chapter XIV

8.5 "Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission, In the Matter of Nebraska Public Power District, Cooper Nuclear Station, Nemaha County, Nebraska, Docket No. 50-298," Issued February 14, 1973.

8.6 NPPD Letter NLS990067 from John H. Swailes (NPPD) to USNRC, dated August 2, 1999, "Withdrawal of Proposed License Amendment, Secondary Containment Isolation Description and Refueling Accident Analysis Results."

8.7 Letter from James R. Hall (USNRC) to Mr. Guy R. Horn (NPPD) dated January 27, 1995, "Cooper Nuclear Station - Amendment No. 167 to Facility Operating License No. DPR-46 (TAC No. M89770)."

8.8. NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal", dated June 3, 1999.

ATTACHMENT 2
DESIGN BASIS ACCIDENT RADIOLOGICAL ASSESSMENT CALCULATIONS
COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

1. Scientech Engineering Calculation 17080-M-01, Rev. 0, X/Q Values for Control Room Intake Using ARCON96, (Cooper Nuclear Station Calculation NEDC 99-031, Attachment 1), Pages 1-90.

2. Scientech Engineering Calculation 17080-M-02, Rev. 0, Control Room Habitability and Offsite Dose for a Fuel Handling Accident , (Cooper Nuclear Station Calculation NEDC 99-032, Attachment 1), Pages 1-161.

3. Scientech Engineering Calculation 17080-M-03, Rev. 0, Control Room, EAB, and LPZ Doses Following a LOCA, (Cooper Nuclear Station Calculation NEDC 99-033, Attachment 1), Pages 1-108.

4. Scientech Engineering Calculation 17080-M-04, Rev. 0, Control Room, EAB, and LPZ Doses Following a CRDA, (Cooper Nuclear Station Calculation NEDC 99-034, Attachment 1), Pages 1-28.

5. Scientech Engineering Calculation 17080-M-05, Rev. 0, Dose Calculation for Control Room, Exclusion Area Boundary, and Low Population Zone for a Main Steam Line Break, (Cooper Nuclear Station Calculation NEDC 99-035, Attachment 1), Pages 1-37.

6. Scientech Engineering Calculation 17080-M-06, Rev. 0, EAB, and LPZ Meteorological Dispersion- Accident Analysis, (Cooper Nuclear Station Calculation NEDC 99-036, Attachment 1), Pages 1-7.

ATTACHMENT 3 LIST OF NRC COMMITMENTS

Correspondence No: NLS 990122

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	