UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 26, 2013

Mr. Oscar A. Limpias
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: REVISIONS TO THE FUEL HANDLING ACCIDENT DESCRIPTION IN THE UPDATED SAFETY ANALYSIS REPORT (TAC NO. ME8992)

Dear Mr. Limpias:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 246 to Renewed Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the CNS Updated Safety Analysis Report (USAR) in response to your application dated June 25, 2012, as supplemented by letter dated March 27, 2013.

The amendment revises the description of the Fuel Handling Accident (FHA) in Section XIV-6.4 of the CNS USAR. The revised USAR FHA description is based on changes to the Design Basis Accident FHA dose calculation, to reflect a 24-month cycle source term using a Global Nuclear Fuels (GNF) 10 x 10 fuel array, a reduced Radial Peaking Factor, and inclusion of a calculated shine contribution to the total dose.

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,

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 246 to DPR-46
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 246
Renewed License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee), dated June 25, 2012, as supplemented by letter dated March 27, 2013, 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, the license is amended by changes to the Cooper Nuclear Station Updated Safety Analysis Report (USAR) and, as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-46 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 246, are hereby incorporated in the license. The licensee 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. Consistent with the requirements in 10 CFR 50.71(e), implementation shall include revision to the Updated Safety Analysis Report, as described in the licensee's application dated June 25, 2012, as supplemented by letter dated March 27, 2013, and the NRC staff's safety evaluation for this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

CFlyon for

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-46
and Technical Specifications

Date of Issuance: June 26, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 246

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Renewed Facility Operating License No. DPR-46 and 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.

Renewed Facility Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
-	--

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2419 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 246, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Cooper Nuclear Station Safeguards Plan," submitted by letter dated May 17, 2006.

NPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NPPD CSP was approved by License Amendment No. 238 as supplemented by a change approved by License Amendment No. 244.

(4) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Cooper Nuclear Station (CNS) Updated Safety Analysis Report and as approved in the Safety Evaluations dated November 29, 1977; May 23, 1979; November 21, 1980; April 29, 1983; April 16, 1984; June 1, 1984; January 3, 1985; August 21, 1985; April 10, 1986; September 9, 1986; November 7, 1988; February 3, 1989; August 15, 1995; and July 31, 1998, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 246 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated June 25, 2012, as supplemented by letter dated March 27, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML121850025 and ML13094A370, respectively), Nebraska Public Power District (NPPD, the licensee) submitted a license amendment request (LAR) to revise the Updated Safety Analysis Report (USAR) for Cooper Nuclear Station (CNS).

The amendment would revise the description of the Fuel Handling Accident (FHA) in Section XIV-6.4 of the CNS USAR. The revised USAR FHA description is based on changes to the Design Basis Accident FHA dose calculation, to reflect a 24-month cycle source term using a Global Nuclear Fuels (GNF) 10 x 10 fuel array, a reduced Radial Peaking Factor, and inclusion of a calculated shine contribution to the total dose.

The supplemental letter dated March 27, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 16, 2013 (78 FR 22570).

2.0 REGULATORY EVALUATION

The NRC staff evaluated the radiological consequences of the postulated design basis accidents (DBAs) against the dose criteria specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," and, using the guidance described in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792). The FHA-specific dose acceptance criteria are specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants:

LWR [Light-Water Reactor] Edition," (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000 (ADAMS Accession No. ML003734190). The dose acceptance criteria for the FHA are a Total Effective Dose Equivalent (TEDE) of 6.3 roentgen equivalent man (rem) at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room (CR) for the duration of the accident. RG 1.183 provides guidance to licensees on acceptable application of alternative source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. The NRC staff also considered relevant information in the CNS USAR.

A revision to 10 CFR 50, Appendix K, "ECCS [Emergency Core Cooling System] Evaluation Models," effective July 31, 2000, allowed licensees to use a power uncertainty of less than 2 percent in design basis loss-of-coolant accident (LOCA) analyses, based on the use of state of the art feedwater flow measurement devices that provide for a more accurate calculation of power. Appendix K did not originally require the power measurement uncertainty be determined, but instead required a 2 percent margin. The revision allows licensees to justify a smaller margin for power measurement uncertainty based on power level instrumentation error. This type of change is also commonly referred to as a measurement uncertainty recapture (MUR) power uprate. The NRC approved a MUR power uprate for CNS by License Amendment No. 231 to Facility Operating License DPR-46 (ADAMS Accession No. ML081540280), dated June 30, 2008.

The NRC approved the implementation of the AST methodology for FHA dose consequence analysis at CNS by License Amendment No. 222 to Facility Operating License DPR-46 (ADAMS Accession No. ML062260239), dated September 5, 2006.

3.0 TECHNICAL EVALUATION

3.1 Proposed USAR Changes

In its letter dated June 25, 2012, the licensee proposed the following changes to the CNS USAR:

3.1.1 Source Term Changes For a 24-Month Fuel Cycle Using a GNF 10 x10 Fuel Array

The current FHA source term is based on the limiting case of GE 14 fuel with a radionuclide inventory based on an 18-month exposure. The transition to a 24-month fuel cycle at CNS will involve the use of a GNF 10x10 fuel array with a radionuclide inventory based on a 24-month exposure. The change in FHA source term is described in USAR Table XIV-6-11 provided in Attachment 2. The resulting dose effects are described in USAR Table XIV-6-16 provided in Attachment 2.

3.1.2 Change to the Radial Peaking Factor

To limit the calculated dose to Control Room occupants that would otherwise increase with the 24-month cycle/GNF 10 x10 source term, the bounding Radial

Peaking Factor was changed from a limit of 2.0 to 1.95. This is described in USAR Section XIV-6.4.7.1 and Table XIV-6-11 provided in Attachment 2.

3.1.3 Inclusion of Shine Contribution

License Amendment 222 accepted a qualitative assessment made by NPPD in the application regarding the potential gamma shine dose from external sources to the Control Room occupants during the FHA (Reference 6.1). That assessment determined that the cloud shine and Control Room Emergency Filter System (CREFS) filter shine contribution to Control Room occupant doses would be a fraction of the inhalation doses and the resulting total dose would still be below regulatory criteria. In the revised FHA dose calculation, NPPD has replaced this qualitative assessment with calculated values that have been added to the dose consequences of the FHA, per Regulatory Guide (RG) 1.183. The new shine contribution is described in USAR Section XIV-6.4.7.4.2 and Table XIV-6-16 provided in Attachment 2.

3.2 NRC Staff Evaluation

3.2.1. Atmospheric Dispersion Estimates

In the Enclosure to the licensee's letter dated June 25, 2012, the licensee stated, in part, that

> The χ/Q values [atmospheric dispersion factors] are taken from existing CNS calculations developed specifically for various Control Room Intake, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) receptor points for use in the development of the bounding Design Basis Accidents (DBA) Radiological Analysis. These receptor locations were previously [determined] to be the most limiting in determining compliance with the dose criteria established.
>
> The control room intake χ/Q values were taken from reference 23 for a release emanating from the Reactor Building. The reactor building vent release case was analyzed as a ground release for three release rates through the reactor building vent. The lowest release flow which coincides with the highest χ/Q values was chosen for the most conservative approach.

Since the input flow rates were not explicitly specified in the June 25, 2012 letter, the NRC issued a request for additional information (RAI) dated March 8, 2013 (ADAMS Accession No. ML13059A345), to confirm the limiting flow rate.

In its RAI response dated March 27, 2013, the licensee confirmed that 1780 cubic feet minute remained the lowest release flow that is appropriate for calculation of the CR χ/Q values and that there were no changes in the FHA release scenario which would alter CNS χ/Q assessments previously performed in support of CNS License Amendment No. 222. Therefore, NRC staff has concluded that the CR, EAB, and LPZ χ/Q values in Table 3.1 below, which are discussed in the safety evaluation associated with License Amendment No. 222, are acceptable for use in the FHA dose assessment related to the current LAR.

3.2.2 Radiological Consequences of a FHA

Consistent with CNS's current licensing basis, the licensee evaluated the dose consequences of an FHA based upon both a 24-hour decay time and a 7-day decay time following reactor shutdown. In its dose calculations, the licensee used the RADionuclide Transport and Removal And Dose Estimation (RADTRAD) computer code, Version 3.03. Secondary containment, secondary containment isolation valves, the standby gas treatment system, or secondary containment isolation instrumentation is not credited after a 24-hour decay period following reactor shutdown. Also, the operability of the Control Room Emergency Filter System (CREFS) and CREFS instrumentation is not credited after a 7-day decay period following reactor shutdown.

The current limiting postulated FHA event assumes a fuel assembly is dropped into the reactor core during refueling operations from a height of 32.95 feet, which is the maximum height allowed by the fuel handling equipment. The resulting impact of the fuel assembly drop onto the top of the core is assumed to damage 150 GNF 10 x10 fuel rods causing a gap release of radionuclides to the water pool above the core. This event could also occur over the spent fuel pool. However, the licensee stated that significantly fewer fuel rods would be damaged in the spent fuel pool drop case, because the maximum drop height in the spent fuel pool is less than 32.95 feet. Since both the reactor cavity and spent fuel pool are located in the reactor building, an FHA in either the reactor cavity or the spent fuel pool is assumed to have the same potential release pathways from the reactor building to the environment. These assumptions are consistent with CNS current licensing basis. Based on the above, the NRC staff concludes that the current limiting postulated FHA event remains applicable for the proposed changes.

For the proposed amendment, the licensee determined the inventory of fission products in the fuel rods and available for release to the containment is based on the maximum full power operation of the core with an assumed core power equal to the current licensed rated thermal power of 2419 Mega-watt thermal (MWt). In addition, this value is multiplied by 1.003977 to account for maximum possible measurement uncertainty as required by Appendix K to 10 CFR 50 for nuclear reactor power operation. The factor of 1.003977 is derived from the current licensed thermal power limit of 2419 MWt and the original Appendix K uncertainty of 2 percent. The NRC staff concludes that the fuel rod fission product inventory calculation is consistent with the regulations in Appendix K and, therefore, is acceptable.

To limit the calculated dose to CR occupants that would otherwise increase with the 24-month cycle GNF 10 x10 source term, the licensee proposed to change the bounding radial peaking factor to a limit of 1.95. The radial peaking factor is applied to the radionuclide inventory to account for differences in power level across the core for a non-LOCA to reflect the maximum possible value as provided by GNF, the fuel vendor. The licensee stated that the maximum expected radial peaking factor per core design would not be expected to exceed 1.7. The radial peaking factor is controlled by CNS Procedure 10.3 of FRED FORM Cycle 27, Rev.1 FORM. The NRC staff concludes that this change is acceptable because the revised value of 1.95 bounds the expected core design value of 1.7. The licensee stated that the combination of the 1.00398 power uncertainty factor applied to the licensed thermal power of 2419 MWt and use of a radial peaking factor of 1.95 results in a conservative source term. The NRC staff also concludes that this statement is acceptable.

The NRC staff accepted a qualitative assessment made by the licensee in the CNS License Amendment No. 222 regarding the potential gamma shine dose contribution from external sources to the CR occupants during the FHA. In the proposed FHA dose calculation, the licensee replaced this qualitative assessment with calculated values that have been added to the dose consequences of the FHA, per RG 1.183. The RADTRAD 3.03 software code was used by the licensee to calculate the revised TEDE doses at the CR receptor location. The revised CR occupant dose also includes gamma shine from both external cloud shine to the CR and CREFS filter shine.

The licensee's FHA evaluation of CR doses for the 24-hour decay time case credited the operability of the CREFS and CREFS instrumentation. During both normal and radiological emergency modes of operation, the CR envelope is positively pressurized and the return air from the CR envelope is recirculated without filtration. During the first minute of the event, the licensee assumed a normal unfiltered inflow of 3235 cubic feet per minute (cfm). The CREFS was then assumed to actuate due to high radiation detected in the reactor building exhaust plenum. For the remaining duration of the event, the licensee assumed an emergency filtered inflow of 810 cfm. The licensee also assumed an unfiltered inleakage of 400 cfm throughout the entire duration of the event. These values are consistent with the CNS current licensing basis.

The licensee's FHA evaluation of CR doses for the 7-day decay time case did not credit the availability of the CREFS. For this scenario, a normal unfiltered inflow of 3635 cfm (which includes 400 cfm inleakage) was assumed for the duration of the accident. The licensee also qualitatively assessed the potential gamma shine dose from external sources to the CR during the FHA event. The radiation sources external to the CR include the airborne external cloud and CREFS filters located within the CR envelope.

RADTRAD was used by the licensee and the output for the activity released to the environment was extracted at the time points consistent with the shine calculation to the CR operators based on a CNS design basis LOCA event. The same methodology and geometry modeling was used in the FHA calculation because the environment geometry model developed for the LOCA calculation is for the same dose point as in this FHA calculation (i.e. the control room personnel). As such, no changes were made in the geometry files, only the source term input files were modified to reflect the FHA RADTRAD source term output.

RADTRAD was also used by the licensee to determine the total amount of activity that was loaded upon the CREFS filter during a FHA release. Also, the licensee assumed higher parameters to be more conservative with regard to the total source term accumulated on the filter. These changes included:

- CREFS Flowrate - increased to 990 cfm versus using 810 cfm. The use of a higher flowrate results in higher halogen accumulation onto the CREFS filter versus the base case. This is conservative as it results in higher shine contribution.

- Filter Efficiency - a value of 100 percent filter efficiency was used for all halogen species as that also maximizes higher halogen accumulation onto the filter versus the base FHA calculation.

The licensee calculated the value of 114 mrem for cloud and CREFS filter CR shine. This value has been added to the dose consequences of the 24-hour decay time case to provide the most limiting dose consequences for the FHA event. The NRC staff concludes that this calculation is acceptable because the methodology and assumptions used are consistent with CNS current licensing basis and the regulatory guidance in RG 1.183.

3.3 NRC Staff Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a FHA and concludes that the licensee has adequately accounted for the effects of the proposed changes to the CNS FHA analysis. The NRC staff further concludes that the plant site and the dose-mitigating engineered safety features remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated TEDE doses at the EAB, LPZ, and in the CR are within regulatory limits. The EPU radiological dose consequences of an FHA are shown in Table 3.2. Therefore, the NRC staff concludes that the licensee's proposed change is acceptable with respect to the radiological consequences of FHA.

Table 3.1

Cooper Fuel Handling Accident Atmospheric Dispersion Factors (sec/m^3)
Ground Level Release from Reactor Building Vent

Time Period	Exclusion Area Boundary	Low Population Zone	Control Room Intake
0-2 hr	5.2×10^{-4}	2.9×10^{-4}	4.15×10^{-3}
2-8 hr	---	2.9×10^{-4}	3.24×10^{-3}
8-24 hr	---	7.3×10^{-5}	1.32×10^{-3}
24-96 hr	---	2.5×10^{-5}	9.01×10^{-4}
96-720 hr	---	5.2×10^{-6}	7.22×10^{-4}

Table 3.2

Calculated FHA Radiological Consequences

	EAB	LPZ	CR
Calculated results, TEDE			
24-hr decay period	1.459	0.809	4.568*
7 day decay period	0.622	0.347	4.393
Dose acceptance criteria, TEDE	6.3	6.3	5

* Includes 114 mrem due to gamma shine from external sources

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 published in the *Federal Register* on April 16, 2013 (78 FR 22570). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

Principal Contributors: D. Duvigneaud
L. Brown

Date: June 26, 2013

June 26, 2013

Mr. Oscar A. Limpias
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
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SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: REVISIONS TO THE FUEL HANDLING ACCIDENT DESCRIPTION IN THE UPDATED SAFETY ANALYSIS REPORT (TAC NO. ME8992)

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Sincerely,

/RA/

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 246 to DPR-46
2. Safety Evaluation

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ADAMS Accession No.: ML13148A225 ***by memo dated May 8, 2013**

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