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

January 31, 2012

Mr. Brian J. O'Grady
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 3.4.3 TO REDUCE THE NUMBER OF SAFETY RELIEF VALVES REQUIRED TO BE OPERABLE FOR OVERPRESSURE PROTECTION (TAC NO. ME5287)

Dear Mr. O'Grady:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 240 to Renewed Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 5, 2011, as supplemented by letters dated October 6 and 18, 2011.

The amendment revises TS 3.4.3, "Safety/Relief Valves (SRVs) and Safety Valves (SVs)." The original proposed TS changes would have revised the required number of safety relief valves (SRVs) required to be operable for overpressure protection and Anticipated Transient without Scram from eight to five. By letter dated October 6, 2011, the licensee revised its submittal to propose revising the required number of SRVs to be operable from eight to seven.

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. 240 to DPR-46
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
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 January 5, 2010, as supplemented by letters dated October 6 and 18, 2011, 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 Technical Specifications 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. 240, 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 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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: January 31, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 240

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
3.4-6	3.4-6

(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. 240, 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.

(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.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) and Safety Valves (SVs)

LCO 3.4.3 The safety function of 7 of 8 SRVs and 3 SVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs or SVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 240 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated January 5, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110100291), as supplemented by letters dated October 6 and 18, 2011 (ADAMS Accession Nos. ML11287A138 and ML11297A033, respectively), Nebraska Public Power District (the licensee), submitted a license amendment request (LAR) in which it requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (CNS).

The amendment would revise TS 3.4.3, "Safety/Relief Valves (SRVs and Safety Valves (SVs)." Specifically, the amendment would reduce the required number of safety relief valves (SRVs) required to be operable for overpressure protection (OPP) and Anticipated Transient without Scram (ATWS) from eight to seven.

The original proposed TS changes would have revised the required number of SRV required to be operable for OPP and ATWS from eight to five. By letter dated October 6, 2011, the licensee supplemented the LAR to propose revising the required number of SRVs to be operable from eight to seven.

The supplemental letters dated October 6 and 11, 2011, 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 November 1, 2011 (76 FR 67488).

2.0 REGULATORY EVALUATION

2.1 System Description (as stated by the licensee)

In its letter dated January 5, 2011, the licensee states:

> CNS is a boiling-water reactor (BWR) of General Electric BWR/4 design, with a Mark I containment. The pressure relief system at CNS includes three Dresser safety valves (SVs) and eight Target Rock SRVs, all of which are located on the main steam lines, within the drywell, between the reactor vessel and the first main steam isolation valve (MSIV). The SVs discharge directly to the interior space of the drywell and the SRVs discharge to the suppression pool.
>
> The safety function (i.e., spring-lift function) of each Target Rock SRV is actuated by a pilot valve assembly. In the event that reactor pressure vessel pressure (RPV) rises to the nominal setpoint (NSP) of the SRV (1080 to 1100 psig [pounds per square inch gauge]), the pilot valve will open, admitting steam and allowing the reactor pressure itself to open the SRV and relieve pressure. The Dresser SVs are held closed by a large mechanical spring. Once the Dresser SV NSP is reached (1240 psig), vessel pressure will act against the spring (directly on the valve disc) to open the SV. Together, the SVs and the spring-lift function of the SRVs provide overpressure protection for the reactor vessel as required by the ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel] Code.
>
> The pilot-actuated, SRV spring-lift function of the Target Rock SRVs is completely independent of the pressure relief function of these valves. In the Automatic Depressurization System (ADS) and Low Low Set (LLS) modes, the SRVs are electro-pneumatically actuated by direct current powered ADS solenoids, which will admit nitrogen from the ADS accumulators to open the SRVs for pressure control. The ADS serves as backup to the High Pressure Coolant Injection system and is actuated by completely different logic/trip signals. The first lift of any SRV coincident with a high-pressure scram signal will arm the LLS function. On second and subsequent lifts, the LLS valves will take over, controlling reactor pressure and minimizing SRV cycling.

In its letter dated January 5, 2001, the licensee also states:

> The safety objective of the pressure relief system is to prevent overpressurization of the nuclear system; this protects the RCPB [reactor coolant pressure boundary] from failure which could result in the uncontrolled release of fission products. In addition, the automatic depressurization feature of the pressure relief system acts in conjunction with the emergency core coolant system for reflooding the core. This protects the reactor fuel cladding from failure due to overheating.

2.2 General Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications,"

which requires that the TSs include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operations (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2)(i), "Limiting conditions for operation," state, in part, that

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulations in 10 CFR 50.36(c)(3), "Surveillance requirements," state that

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

CNS construction predated the issuance of 10 CFR 50, Appendix A[1], "General Design Criteria for Nuclear Power Plants." CNS is designed to conform to the proposed general design criteria (GDC) published in the July 11, 1967, *Federal Register*, except where commitments were made to specific 1971 GDC. The Atomic Energy Commission accepted CNS conformance with the proposed GDC. By letter dated January 5, 2011, the licensee appropriately identified the following draft GDC as specified in Appendix F to the CNS Updated Safety Analysis Report (USAR):

- USAR Appendix F, Criterion 9, Reactor Coolant Pressure Boundary

  The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- USAR Appendix F, Criterion 14, Core Protection Systems

  Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

- USAR Appendix F, Criterion 37, Engineered Safety Features Basis for Design

  Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered

[1] The 1967 Proposed GDC as described in the CNS updated safety analysis report, Appendix F, are the licensing basis for CNS; however, the NRC staff concluded in its 1973 Safety Evaluation Report for CNS that the intent of the 1971 Final Rule for 10 CFR Part 50, Appendix A, had also been met.

> safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes

The current CNS TS 3.4.3 LCO states,

> The safety function of 8 SRVs and 3 SVs shall be OPERABLE.

Revised TS 3.4.3 LCO would state,

> The safety function of 7 of 8 SRVs and 3 SVs shall be OPERABLE.

In addition, the current TS 3.4.3 LCO Condition A. states,

> One or more SRVs or SVs inoperable.

Revised TS 3.4.3 LCO Condition A. would state,

> One or more required SRVs or SVs inoperable.

The licensee also proposed to make conforming changes to the TS Bases.

3.2 Licensee Evaluation

As part of the analysis of the proposed reduction of number of SRVs required, GE Hitachi submitted analysis of ASME OPP and ATWS. These evaluations were performed with three SRVs out of service (OOS). After the supplement to propose one SRV OOS, the results of three SRVs OOS are still bounding.

The limiting event for the ASME OPP is the MSIV Closure on High Flux. The analysis conservatively assumed core power as well as core flow. End of Cycle conditions were also assumed and are most limiting due to power shapes typically being peaked towards the top of the core and the worth of the early part of the scram is minimized. SRV setpoints were varied and conservatively assumed to be 3 percent above the setpoint.

The results of the ASME OPP analysis showed that with three SRVs OOS, the remaining SRVs would be able to mitigate the OPP event with margin to both the dome pressure safety limit of 1337 psig and the peak vessel pressure of 1375. These results conservatively bound the supplemental LAR of one OOS SRV.

GE Hitachi used the bounding ATWS evaluation to determine the maximum number of SRVs that could be OOS and still mitigate the event. The evaluation showed that three SRVs could be out of service and the ATWS could still be mitigated. The original LAR requested three SRVs OOS stemming from this evaluation of the maximum number of SRVs OOS in the most limiting event.

Peak reactor power was assumed and both Beginning of Cycle and End of Cycle conditions were evaluated. The SRVs were assumed to open at the +3 percent range of their setpoint as well. Two ATWS events were analyzed; ATWS with Main Steam Isolation Valve Closure and with Pressure Regulator Failure – Open. These two events were evaluated since they were found to be substantially more limiting than ATWS with Loss of Auxiliary Power or with Inadvertent Opening of Relief Valve.

The results showed that CNS would stay under the allowed limits for the ATWS events. The peak vessel pressure was shown to stay below 1500 psig. The peak cladding temperature stayed below 2200 degrees Fahrenheit. The peak vessel pressure came out to be 1489 psig and the peak cladding temperature came out to be 1442 degrees Fahrenheit. These results were under the limits and also came from cases with three SRVs OOS.

3.3. NRC Staff Evaluation

By letter dated January 5, 2011, NPPD requested that three SRVs be allowed to be OOS. In evaluating the proposed change, the NRC staff determined that additional information was needed to complete its review (ADAMS Accession No. ML111430704). Among other issues, the NRC questioned whether the proposed change would involve a significant increase in the probability or consequences of an accident previously evaluated and involve a significant reduction in a margin of safety as described in the Updated Safety Analysis Report. During a publicly noticed telephone call on July 12, 2011 (ADAMS Accession No. ML112170081), NPPD stated that it would submit a revision to the LAR revising the No Significant Hazards Consideration (NSHC) clarifying why the change in probability or consequences and margin is not significant. Subsequent to the public telephone call, by letter dated October 6, 2011, NPPD decided to supplement the LAR to request seven of eight SRVs be required rather than five of eight SRVs and submitted a revised NSHC.

The NPPD request would allow one SRV to be OOS. The NRC staff concludes that the licensee has shown that there is reasonable assurance of safety with one SRV OOS and that sufficient margin is maintained throughout the OPP and ATWS events, as evaluated.

3.4 NRC Staff Conclusion

The NRC staff has reviewed the proposed changes to TS 3.4.3 regarding the reduction of the safety function of SRVs required to be operable from eight to seven. The staff concludes that the licensee provided sufficient analysis and calculations to show that CNS would be reasonably assured to operate safely and that margin is maintained in OPP and ATWS scenarios. Furthermore, the NRC staff concludes that the proposed TS changes are in accordance with 10 CFR 50.36 and the applicable draft GDC as described in the CNS USAR. Therefore, the NRC staff concludes that the proposed TS change is acceptable.

The NRC staff has no objection to the licensee's proposed changes to the TS Bases.

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 November 1, 2011 (76 FR 67488). 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) 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 Contributor: J. Miller

Date: January 31, 2012

January 31, 2012

Mr. Brian J. O'Grady
Vice President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION 3.4.3 TO REDUCE THE NUMBER OF SAFETY RELIEF VALVES REQUIRED TO BE OPERABLE FOR OVERPRESSURE PROTECTION (TAC NO. ME5287)

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Sincerely,
/RA/

Lynnea E. Wilkins, Project Manager
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Division of Operating Reactor Licensing
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