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

November 12, 2009

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

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH TESTING (TAC NO. ME1388)

Dear Mr. Minahan:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 2, 2009.

The amendment would (1) delete TS surveillance requirement (SR) 3.1.3.2 and revise SR 3.1.3.3, (2) remove reference to SR 3.1.3.2 from Required Action A.3 of TS 3.1.3, "Control Rod OPERABILITY," and (3) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action."

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,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 235 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 FACILITY OPERATING LICENSE

Amendment No. 235
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 2, 2009, 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 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. 235, 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

James R. Hall for

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

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

Date of Issuance: November 12, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 235

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the 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.

Facility Operating License

REMOVE	INSERT
Page 3 of 5	Page 3 of 5

Technical Specifications

REMOVE	INSERT
1.4-4	1.4-4
1.4-5	1.4-5
3.1-8	3.1-8
3.1-10	3.1-10

(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. 235, 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.

(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.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE-------------------------- Not required to be performed until 12 hours after $\geq$ 25% RTP. --- Perform channel adjustment.	 7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches $\geq$ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power $\geq$ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be met in MODE 1. ---	
Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM.
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	-----------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -------------------------------- Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	(Deleted)	
SR 3.1.3.3	------------------------------NOTE------------------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -- Insert each withdrawn control rod at least one notch.	 31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 TO

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By application dated June 2, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091560032), Nebraska Public Power District (the licensee), requested changes to the Technical Specifications (TSs) for Cooper Nuclear Station (CNS). The proposed changes would (1) delete TS surveillance requirement (SR) 3.1.3.2 and revise SR 3.1.3.3, (2) remove reference to SR 3.1.3.2 from Required Action A.3 of TS 3.1.3, "Control Rod OPERABILITY," and (3) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

The changes are in accordance with U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action." TSTF-475, Revision 1, was approved for use by the NRC on November 5, 2007 (ADAMS Accession No. ML073050017), and made available to the industry by the NRC on November 13, 2007 (72 FR 63935), through the consolidated line item improvement process (CLIIP).

TSTF-475 revised the reference Standard Technical Specifications (STS) by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from 7 days after the control rod is withdrawn and THERMAL POWER is greater than the Low Power Setpoint (LPSP) of the Rod Worth Minimizer (RWM) to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM," and (2) revising Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column. NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," is the STS for CNS, which is a boiling-water reactor (BWR)-4.

The licensee stated in its application that it is proposing two variations, as described below in the technical evaluation, from the applicable TS changes described in the modified TSTF-475, Revision 1. The licensee also stated that the justifications presented in the TSTF and the NRC staff safety evaluation for the TSTF are applicable to CNS.

The purpose of the surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) mechanism (CRDM), by which the control rods are moved, are components of the CRD system (CRDS), which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable, with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event, and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the change to Example 1.4-3 in Section 1.4, "Frequency," is to clarify the applicability of the 25 percent allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column.

2.0 REGULATORY EVALUATION

In Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36), the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

As stated in 10 CFR 50.36(c)(2)(i), LCOs 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 ..." The remedial actions in the TSs are specified in terms of LCO conditions, required actions, and completion times (CTs), or allowed outage times (AOTs), to complete the required actions. When an LCO is not being met, the CTs specified in the TSs are the time allowed in the TSs for completing the specified required actions. The conditions and required actions specified in the TSs must be acceptable remedial actions for the LCO not being met, and the CTs must be a reasonable time for completing the required actions while maintaining the safe operation of the plant.

As required by 10 CFR 50.36(c)(2)(ii), an LCO must be included in TSs for any item meeting one of the following four criteria:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Those items that do not fall within or satisfy any of the above criteria are not required to be included in the TSs.

As required by 10 CFR 50.36(c)(3), SRs are the requirements related 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 LCOs will be met.

The regulations in 10 CFR, Part 50, Appendix A, General Design Criterion (GDC) 26, "Reactivity control system redundancy and capability,"[1] state that:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOOs], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29, "Protection against anticipated operational occurrence," states that

> The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The design relies on the control rod drive CRDS to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine

[1] The 1967 Proposed GDC as described in the CNS updated safety analysis report (USAR), 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.

generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. Compliance with GDCs 26 and 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

As discussed in the technical evaluation below, the proposed changes still assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

3.0 TECHNICAL EVALUATION

3.1 Revise the SR Frequency for Notch Testing of Each Fully Withdrawn Control Rod from Weekly to Monthly (TS 3.1.3, "Control Rod OPERABILITY")

The CRDS consists of CRDs, which are hydraulically-operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed (e.g., a 360-degree break due to an intergranular stress-corrosion cracking (IGSCC)-initiated crack), and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results, but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The hydraulic control units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Reference 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Reference 1), and since the CRT crack growth rate is slow (Reference 1), the applicant maintains that it should be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Reference 1), based on fundamental principles of stress-corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization, and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff concludes that increasing the surveillance interval from 7 days to 31 days would not compromise the CRDS's capability to reliably control reactivity changes under normal operations, including AOOs, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Reference 1). The staff's conclusions are not based upon implementation of either of these recommendations; rather, they are based on operational experience, slow crack growth, and potential safety benefits of reduced control rod movements. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

The licensee has proposed to delete SR 3.1.3.2 for the fully withdrawn control rods and to remove the word "partially" from SR 3.1.3.3 for the partially withdrawn control rods. By removing the word "partially," SR 3.1.3.3 will apply to any withdrawn control rod. The revised SR 3.1.3.3 would then apply to both fully withdrawn and partially withdrawn control rods.

Therefore, based on the NRC staff's review of the above proposal, the NRC staff concludes that (1) the deletion of SR 3.1.3.2 by combining SRs 3.1.3.2 (fully withdrawn controls) and 3.1.3.3 (partially withdrawn control rods) into one surveillance for any withdrawn control rod, and the extension of the frequency to 31 days for the fully withdrawn control rods is acceptable.

Deleting SR 3.1.3.2 from TS 3.1.3 Required Action A.3

The licensee proposes to delete SR 3.1.3.2, and replace it with the word "deleted." Deleting SR 3.1.3.2 from Required Action A.2 does not delete any requirements from the TSs, because SR 3.1.3.3 now applies to both fully withdrawn and partially withdrawn control rods.

Based on the above, the NRC staff concludes that the proposed change is acceptable.

3.2 Clarify in TS Example 1.4-3 that the 1.25 Surveillance Test Interval Extension in SR 3.0.2 is Also Applicable to Time Periods Discussed in SR Notes

Regarding the change to Example 1.4-3 in Section 1.4, "Frequency," this change makes it clear that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the "FREQUENCY" column and in Notes in the "SURVEILLANCE" column. This change to Example 1.4-3 is linked to TSTF-475 since SR 3.1.3.3 contains a 31-day time period in both the "SURVEILLANCE" column and in the "FREQUENCY" column, and the revised Example makes it clear that the 1.25 provision is equally applicable to both of these 31-day periods in SR 3.1.3.3. This change is proposed to be consistent with the definition of "specified Frequency" provided in the second paragraph of Section 1.4. This paragraph states:

> The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

As made clear in the second sentence above, the "specified Frequency" includes time periods discussed in Notes in the "Surveillance" column, in addition to time periods listed in the "Frequency" column. Therefore, the provisions of SR 3.0.2 (which permit a 25 percent grace period to facilitate surveillance scheduling and avoid plant operating conditions that may not be suitable for conducting the test) also apply to the time periods listed in Notes in the "SURVEILLANCE" column. This is because SR 3.0.2 states that "[t]he *specified Frequency* (emphasis added) for each SR is met if the Surveillance is performed within 1.25 times the interval specified...."

Therefore, the licensee proposes to revise Example 1.4-3 to be consistent with the above statements. The example currently explicitly recognizes that the 25 percent extension allowed by SR 3.0.2 is applicable to the time period listed in the "FREQUENCY" column, but it does not explicitly recognize that the SR 3.0.2 extension is applicable to the time period listed in the NOTE in the "SURVEILLANCE" column. The change to the Example provides this explicit recognition by copying the phrase "(plus the extension allowed by SR 3.0.2)" in two additional portions of the discussion for this Example.

Based on the above, the NRC staff concludes that the proposed addition of the phrases to Example 1.4-3 of the CNS TSs meets the requirements of 10 CFR 50.36 and is acceptable.

3.3 Clarify the Requirement to Fully Insert All Insertable Control Rods for TS 3.3.1.2, "Source Range Monitoring (SRM) Instrumentation," Required Action E.2

The licensee did not propose this change because the word "fully" is already contained in CNS TS 3.3.1.2, Required Action E.2. The NRC staff agrees that this change is not needed for CNS.

The licensee proposed two deviations from the TS changes described in TSTF-475-A, Revision 1. The first deviation is to reflect not renumbering the remaining SRs after deleting SR 3.1.3.2. The licensee states in its application that this eliminates the need to revise Table 3.1.4-1. The second deviation is in the TS Bases for SR 3.1.3.3 to reflect just the single SR 3.1.3.3 and to apply the potential power reduction basis to all withdrawn control rods rather than just those partially withdrawn. The licensee stated that these deviations conform the Bases to the SR. The NRC staff concludes that the licensee's proposed deviations do not affect the applicability of the safety evaluation and are therefore acceptable.

3.4 Conclusion

Based on its evaluation above of the proposed changes, the NRC staff finds these changes meet the requirements of 10 CFR 50.36 and are acceptable. The staff also concludes that the proposed TS revisions will have a minimal effect on the high reliability of the CRDS, while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Thus, the plant continues to meet the requirements of GDCs 26 and 29. Since the amendment request meets the requirements of GDCs 26 and 29 and 10 CFR 50.36, the NRC staff concludes that the proposed changes are acceptable.

4.0 REGULATORY COMMITMENT

The licensee made the following regulatory commitment in its application:

> Nebraska Public Power District will establish the Technical Specification Bases changes for TS B3.1.3 consistent with those shown in TSTF-475-A, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," with the variations noted.

The licensee scheduled the commitment to be implemented with implementation of this amendment. The NRC staff concludes that the regulatory commitment is acceptable.

5.0 STATE CONSULTATION

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

6.0 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 and changes a surveillance requirement. 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 June 30, 2009 (74 FR 31325). 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.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. BWR Owners Group to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," BWROG-06036, dated November 16, 2006, with Enclosure of the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," dated November 2006, ADAMS Accession No. ML063250258.

1. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0, 'Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,' dated February 28, 2007," (TSTF-475 Revision 1 is an enclosure) TSTF-07-19, dated May 22, 2007, ADAMS Accession No. ML071420428.

Principal Contributor: R. Grover

Date: November 12, 2009

November 12, 2009

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

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH TESTING (TAC NO. ME1388)

Dear Mr. Minahan:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 235 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 2, 2009.

The amendment would (1) delete TS surveillance requirement (SR) 3.1.3.2 and revise SR 3.1.3.3, (2) remove reference to SR 3.1.3.2 from Required Action A.3 of TS 3.1.3, "Control Rod OPERABILITY," and (3) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action."

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/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures:
1. Amendment No. 235 to DPR-46
2. Safety Evaluation

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