October 26, 2001

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

SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT TO REVISE THE TECHNICAL SPECIFICATIONS SURVEILLANCE TEST REQUIREMENT SR 3.6.1.3.8, FOR EXCESS FLOW CHECK VALVES (EFCVs) (TAC NO. MB1820)

Dear Mr. Wilson:

The Commission has issued the enclosed Amendment No. 189 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated April 12, 2001. Your application also requested the approval of Inservice Testing (IST) relief request number RV-10, which is being reviewed separately.

CNS TS surveillance requirement (SR) 3.6.1.3.8, currently requires verification of the actuation capability of each reactor instrumentation line EFCV every 18 months, to demonstrate that each reactor instrumentation line EFCV is operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break. The amendment revises TS SR 3.6.1.3.8, to relax the 18-month EFCV surveillance frequency by limiting the number of tests to a "representative sample" every 18 months such that each EFCV will be tested at least once every 10 years. The amendment adopts the NRC staff's approved Technical Specifications Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000.

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/

Girija S. Shukla, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-298

Enclosures: 1. Amendment No. 189 to DPR-46
2. Safety Evaluation

cc w/encls: See next page

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DATE	10/24/01	10/24/01	10/24/01	09/21/01	09/27/01	10/24/01

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cc:

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

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

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

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

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

Mr. Michael J. Linder, Director
Nebraska Department of Environmental
Quality
P. O. Box 98922
Lincoln, NE 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, NE 68305

Ms. Cheryl K. Rogers, Program Manager
Nebraska Health & Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P. O. Box 95007
Lincoln, NE 68509-5007

Mr. Ronald A. Kucera, Director
of Intergovernmental Cooperation
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65102

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 218
Brownville, NE 68321

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Jerry Uhlmann, Director
State Emergency Management Agency
P. O. Box 116
Jefferson City, MO 65101

Chief, Radiation Control Program, RCP
Kansas Department of Health
and Environment
Bureau of Air and Radiation
Forbes Field Building 283
Topeka, KS 66620

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.189
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated April 12, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, 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. 189, are hereby incorporated in the license. The Nebraska Public Power District shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 26, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 189

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

REMOVE	INSERT
3.6-14	3.6-14
B 3.6-27	B 3.6-27

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By a letter dated April 12, 2001, Nebraska Public Power District (licensee) submitted a request for amendment to license No. DPR-46 for Cooper Nuclear Station (CNS) and a request for relief from inservice testing (IST) requirements. The relief request is being handled by the NRC staff as a separate action. The license amendment request involves the licensee's request to revise the Technical Specification (TS) surveillance requirement (SR) 3.6.1.3.8.
TS SR 3.6.1.3.8 currently requires verification of the actuation capability of each reactor instrumentation line excess flow check valve (EFCV) every 18 months, to demonstrate that each reactor instrumentation line EFCV is operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break.

The licensee's proposed TS change would revise TS SR 3.6.1.3.8 to relax the 18-month EFCV surveillance frequency by limiting the number of tests to a "representative sample" every 18 months, such that each EFCV will be tested at least once every 10 years. The proposed change is consistent with the approved Technical Specifications Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000.

2.0 BACKGROUND

EFCVs are installed in boiling water reactor (BWR) instrument lines penetrating the primary containment boundary to limit the release of fluid in the event of an instrument line break. Regulatory Guide (RG) 1.11, "Instrument Lines Penetrating Primary Reactor Containment," provides guidance on the implementation of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 55, "Reactor Coolant Pressure Boundary Penetrating Containment," and GDC 56, "Primary Containment Isolation," for instrumentation lines that penetrate primary reactor containment and are part of the reactor coolant pressure boundary. As stated in RG 1.11, EFCVs in combination with flow restricting features (line size or orifice) satisfy the requirements of GDC 55 and 56 for automatic isolation capability, maintain the reliability of the connected instrumentation, and ensure the functional performance of secondary containment in the event of an instrumentation line rupture. Examples of EFCV installations include reactor pressure vessel level and pressure instrumentation, main steam line flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steam line flow instrumentation. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post loss-of-coolant accident conditions.

CNS TS SR 3.6.1.3.8 currently requires verification of the actuation capability of each reactor instrumentation line EFCV every 18 months. The SR demonstrates that each reactor instrumentation line EFCV is operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break. The proposed change revises TS SR 3.6.1.3.8 to relax the 18-month EFCV surveillance frequency by limiting the number of tests to a “representative sample” every 18 months, such that each EFCV will be tested at least once every 10 years (nominal). The “representative sample” consists of approximately equal numbers of EFCVs being tested every 18 months such that each EFCV is tested at least once every 10 years.

The basis for the request is the high degree of reliability shown by the EFCVs and the low consequences of an EFCV failure. The supporting analysis for the licensee’s conclusion is based on General Electric Nuclear Energy (GENE) Topical Report NEDO-32977-A, “Excess Flow Check Valve Testing Relaxation,” dated June 2000. The topical report provided: (1) an estimate of steam release frequency into the reactor building due to a break in an instrument line concurrent with an EFCV failure to close and (2) an assessment of the radiological consequences of such a release. The Boiling Water Reactor Owners’ Group (BWROG) concluded that EFCVs testing intervals could be extended up to 10 years based on the topical report reliability and consequence analysis without significantly affecting plant risk. The BWROG suggested a staggered test interval based on actual valve performance with each valve being tested at least once every 10 years. The NRC staff accepted the generic applicability of the topical report by a safety evaluation report (SER) dated March 14, 2000, and agreed that the EFCV test interval could be extended to as much as 10 years. The NRC staff also noted that licensees adopting the topical report must have a failure feedback mechanism and corrective action program to ensure that EFCV performance continues to be bounded by the topical report results. Additionally, each licensee is required to perform a plant-specific radiological dose assessment and EFCV failure rate and release frequency analysis to confirm that their facility is bounded by the generic analysis of the topical report.

The proposed change adopts the NRC staff’s approved TSTF-334, Revision 2. TSTF-334 was approved by the NRC staff on October 31, 2000, by letter from W. D. Beckner to A. R. Pietrangelo (NEI). It proposed specific changes to the Standard Technical Specifications (STS) providing guidance for licensees implementing the extended EFCV surveillance test intervals proposed in the topical report. TSTF-334 is applicable only for those plants for which NEDO-32977-A is applicable and which are subject to EFCV performance and corrective action criteria.

3.0 EVALUATION

The NRC staff reviewed the licensee’s submittal for conformance to the March 14, 2000, NRC staff SER to Topical Report NEDO-32977-A and the guidance of approved TSTF-334 Revision 2. The NRC staff’s evaluation focused on the following areas: (1) EFCV failure rate and release frequency, (2) the licensee’s failure feedback mechanism and corrective action program, (3) radiological dose assessment, and (4) conformance of the revised TS to generic TS guidance.

3.1 EFCV Failure Rate and Release Frequency

In the topical report, EFCV reliability was evaluated based on testing experience provided by 12 different BWR plants. The composite data indicated that EFCVs are very reliable. The data represented 12,424.5 valve years of operation, with a total of 11 failures noted. The EFCV composite failure rate was 1.67E-07/hour and was referenced as the "upper limit" failure rate in the topical report.

The NRC staff noted in its review of the report that the BWROG assumed that the EFCV failure rate was constant over time and did not account for potential age-related degradation in the EFCV failure rate. Additionally, the NRC staff questioned the use of an instrument line break frequency based on WASH-1400 and not on more current data. To address this concern, the BWROG Request for Additional Information (RAI) response included an updated instrument line failure frequency of 3.52E-05 failures/year based on the Electric Power Research Institute's Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," dated July 1992. This value is 6.6 times greater than the value calculated in the topical report using WASH-1400 data. The BWROG RAI response also assumed that the observed EFCV failures were 5 times the actual observed number (55 vs.11) listed in the topical report. The additional impact of an increase in instrument line failure frequency and a fivefold increase in EFCV failures assumed by the BWROG RAI response demonstrated that release frequencies remained low with limited impact.

To estimate the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency downstream of the EFCV, and (2) the probability of the EFCV failing to close. The CNS data were found to be consistent both in time sampled and EFCV reliability (zero EFCV failures, 68 valves per unit and 680 valve years operating time) when compared to the topical report data. Using the current CNS surveillance interval of 18 months, an instrument line break frequency of 2.39E-03/year based on 68 valves installed at CNS, and a total plant EFCV failure frequency of 5.53E-03/year, the CNS EFCV release frequency is estimated to be 9.91E-06/year. For a surveillance interval of 10 years, the release frequency is estimated to be 6.61E-05/year. The 10-year release frequency shows an increase of 5.62E-05/year over the 18-month value. This represents the increase in the total plant release frequency for a random break of any of the CNS 68 instrument lines with a concurrent failure of the EFCV to isolate the break. These values are consistent with the NRC staff topical report SER, which concluded that an increase in release frequency of 7.3E-05/year was not significant. The CNS plant specific EFCV failure and release rates are also comparable with industry data and the results given in the topical report. Based on the above, the NRC staff does not consider the estimated increase in release frequency for CNS to be significant.

3.2 Licensee's Failure Feedback Mechanism and Corrective Action Program

The NRC staff noted that the topical report does not provide a specific failure feedback mechanism, but does state that a plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The BWROG responded to the NRC staff RAI question concerning failure feedback by stating that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism responsive to EFCV failure trends is in place.

The licensee stated that the CNS 10 CFR 50.65 Maintenance Rule Program will be revised to provide a means to track the performance of the EFCVs. To ensure EFCV performance remains consistent with the extended test interval, minimum performance criteria have been established by the licensee. The criteria for reactor instrument line EFCVs have been established for CNS as less than or equal to 2 functional failures on a 36-month rolling average to ensure that the EFCV performance remains consistent with the extended surveillance interval assumptions and adverse trends in EFCV performance are identified. The NRC staff considers the licensee's program to be acceptable because it accounts for potential changes in EFCV failure rates and satisfies TSTF-334 performance and corrective action criteria.

3.3 Radiological Dose Assessment

The licensee referred to the original licensing Safety Evaluation, dated February 14, 1973, to show that the radiological consequences of an instrument line break have previously been considered in accordance with Regulatory Guide 1.11 and found acceptable to the NRC staff. For the previous analysis, the 1/4-inch orifices were credited to prevent overpressurization of the reactor building and limit offsite doses to values substantially below the 10 CFR Part 100, "Reactor Site Criteria," values. The radiological consequences for an instrument line break are not impacted by the proposed TS surveillance changes since there is no change in the function or operation of the restricting orifices to limit the blowdown, and the proposed changes do not cause a change to the radiological source term. Since there are no changes to the analysis assumptions, the offsite radiological consequences of an instrument line break remain substantially below the guidelines of 10 CFR Part 100.

The licensee also performed a radiological sensitivity evaluation by comparing CNS plant-specific values to those in the BWROG Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," Attachment B, "Instrument Line Break Radiological Analysis," and then calculating the resulting offsite doses. The licensee compared plant-specific values with those stated in the BWROG report Attachment B for (1) offsite atmospheric dispersion factors, (2) number of fuel bundles in the core, (3) the mass of reactor pressure vessel water, and (4) CNS technical specification dose equivalent Iodine-131 values. All other parameters were as stated in the BWROG report attachment and were found to generally bound the plant-specific values. It should be noted that the atmospheric dispersion factors used in the sensitivity analysis are currently under review by the NRC staff for another licensing action. Approval of the proposed EFCV TS changes does not imply approval of the atmospheric dispersion factors, but is based on maintaining the current licensing basis as verified by the sensitivity analysis results. However, the atmospheric dispersion factors are on the same order of magnitude as those used for the original licensing of the plant; therefore, the NRC staff has determined that the licensee's sensitivity evaluation is reasonable. The NRC staff finds that the sensitivity evaluation shows that the conclusions in the BWROG report are applicable to CNS. The NRC staff also finds that the licensee's sensitivity evaluation confirms that the current CNS licensing basis remains acceptable, with offsite radiological consequences of an instrument line break substantially below the guidelines of 10 CFR Part 100.

Based on the above, the NRC staff agrees with the licensee's determination that the current licensing basis remains applicable for the proposed EFCV surveillance interval, with regard to the potential radiological consequences of an instrument line break with failure of the EFCV to isolate.

3.4 Conformance of the Proposed TS to Generic TSTF Guidance

The CNS TS SR 3.6.1.3.8, currently requires verification that each reactor instrumentation line EFCV be demonstrated OPERABLE at least once every 18 months by verifying the valve actuates to the isolation position on an actual or simulated instrument line break. The current sentence in TS SR 3.6.1.3.8, will be revised to read, “Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break.”

The term “representative sample,” as proposed by the topical report and TSTF-334, is not defined in the TS itself. However, the BWROG response to the NRC staff RAI stated that the term “representative sample,” with an accompanying explanation in the TS Bases, is identical to the current usage in the STS, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term “representative” in TS SR 3.8.6.3 in reference to battery cell testing, and “representative sample” in SR 3.1.4.2 for verification of control rod scram times. The criteria for “representative sample” and the basis for the nominal 10-year testing interval are provided in the licensee submittal, and are similar to Insert 1 and Insert 2 provided in the NRC staff’s approved TSTF-334, Revision 2. Therefore, the application of a “representative sample” for the EFCV testing SR, with an accompanying explanation in the TS Bases, is consistent with TSTF-334, Revision 2 to the STS usage and is therefore acceptable to the NRC staff.

The licensee included in its submittal, for information, a revised Bases for SR 3.6.1.3.8, that includes a discussion of the EFCV test frequency and the term “representative sample.” The Bases for SR 3.6.1.3.8 includes the following insert:

> “This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that each valve actuates to the isolation position on an actual or simulated instrument line break. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event. The 18-month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.
>
> The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.”

In addition, the NRC staff reviewed the revised TS wording in SR 3.6.1.3.8, and finds the proposed revision to be consistent with TSTF-334 and TS generic guidance.

3.5 Conclusion

As demonstrated in GENE Topical Report NEDO-32977-A, the impact of an increase in the EFCV surveillance test interval to 10 years results in an instrument line release frequency that is considered by the NRC staff to be sufficiently low, especially since the consequences of an EFCV failure are bounded by previous licensee analysis; and therefore, are highly unlikely to lead to core damage. Additionally, the licensee's evaluation results (including the plant specific EFCV failure data and release frequency) is consistent with the topical report composite results. The NRC staff concludes that the release frequency associated with the CNS request for relaxation of ECFV surveillance testing is sufficiently low and therefore acceptable.

The consequences of steam release from the failure of the EFCVs is not significant, as shown by the previous licensee analysis. Based on the acceptability of the methods applied to estimate the release frequency, the licensee's relatively low release frequency estimate, the negligible consequences of a release in the reactor building, in conjunction with a highly unlikely impact on core damage, the NRC staff concludes that the impact on risk associated with the CNS request for relaxation of ECFV surveillance testing is also sufficiently low and therefore acceptable.

The topical report established that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. These programs ensure that meaningful feedback data are acquired so that appropriate corrective action may be taken with regard to EFCV performance. The licensee provided information to the NRC staff regarding EFCV performance criteria and the EFCV corrective action program. The NRC staff finds the licensee's program to be in conformance with TSTF-334, Revision 2, and the topical report, and thus acceptable.

Based on the above, the NRC staff finds the proposed change to relax the CNS instrument line EFCV surveillance frequency, by allowing a representative sample of EFCVs to be tested every 18 months, with all EFCVs being tested at least once every 10 years (nominal), to be acceptable. The proposed change conforms with TSTF-334 generic guidance, Topical Report NEDO-32977-A, and the NRC staff's safety evaluation of March 14, 2000.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 48289). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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.

Contributors: C. Doutt and N. Le

Date: October 26, 2001