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NLS2007048
August 16, 2007

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: License Amendment Request for a One-Time Exception to the Five-Year Test Frequency for a Single Safety Valve
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Dear Sir or Madam,

The purpose of this letter is for the Nebraska Public Power District (NPPD) to request an amendment to Facility Operating License DPR-46 in accordance with the provisions of 10 CFR 50.4 and 10 CFR 50.90 to revise the Cooper Nuclear Station (CNS) Technical Specifications (TS). This request revises TS Section 5.5.6, Inservice Testing Program to allow a one-time exception of the five-year frequency requirement for setpoint testing of safety valve MS-RV-70ARV.

The five-year interval for this safety valve expires on March 10, 2008, during the current operating cycle (Cycle 24). This request is based on the need to perform testing of this valve when shutdown. The next refueling outage is scheduled to begin in April 2008. The requested one-time exception involves an extension of 90 days from the current due date of March 10, 2008, to June 8, 2008. The surveillance requirement (SR) being extended is SR 3.4.3.1 as it specifically pertains to safety valve MS-RV-70ARV.

NPPD requests Nuclear Regulatory Commission (NRC) approval of the proposed TS change and issuance of the requested license amendment by February 8, 2008. Approval by that date is needed to avoid the need to start planning to shut down CNS on or before March 10, 2008. Failure to obtain the requested amendment prior to March 10, 2008 would require an unnecessary shutdown of CNS. The amendment will be implemented within 30 days of issuance of the amendment. The one-time extension proposed in this amendment request expires upon shut down in the next refueling outage.

Attachment 1 provides a description of the proposed TS change, the technical analysis basis for the change, the no significant hazards consideration evaluation pursuant to 10 CFR 50.91(a)(1), and the environmental impact evaluation pursuant to 10 CFR 51.22. Attachment 2 provides

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marked up pages with the specific changes to the current CNS TS. Attachment 3 provides the revised TS pages in final format. No TS Bases pages are affected by this amendment request.

This proposed TS change has been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board). Amendments to the CNS Facility Operating License through Amendment 226 dated October 31, 2006, have been incorporated into this request. This request is submitted under oath pursuant to 10 CFR 50.30(b).

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

Should you have any questions concerning this matter, please contact David Van Der Kamp, Acting Licensing Manager, at (402) 825-2904.

I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 8/16/07
Date

Sincerely,

Stewart B. Minahan
Vice President – Nuclear and
Chief Nuclear Officer

/rr

Attachments

cc: Regional Administrator w/ attachments
USNRC - Region IV

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

Senior Resident Inspector w/ attachments
USNRC - CNS

Nebraska Health and Human Service w/ attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/ attachments

ATTACHMENT 1

LICENSE AMENDMENT REQUEST FOR A ONE-TIME EXCEPTION TO THE FIVE-YEAR TEST FREQUENCY FOR A SINGLE SAFETY RELIEF VALVE

COOPER NUCLEAR STATION
DOCKET NO. 50-298, DPR-46

Revised Technical Specification Page

5.0-10

1.0 Description

This license amendment request (LAR) proposes a one-time extension of setpoint testing of a single safety valve at the Nebraska Public Power District (NPPD) Cooper Nuclear Station (CNS). This extension would be allowed by addition of an exception to the provision that prohibits application of the 25 percent extension of surveillance intervals of Surveillance Requirement (SR) 3.0.2 in Technical Specifications (TS) Section 5.5.6, Inservice Testing Program. The requested extension will extend the setpoint testing out to the next scheduled refueling outage, but no later than June 8, 2008.

2.0 Proposed Change

This LAR proposes to revise TS Section 5.5.6, Inservice Testing Program, by adding a subparagraph to allow a one-time exception of the five-year frequency requirement for setpoint testing of safety valve (SV) MS-RV-70ARV in order to coincide with the Refueling Outage (RFO) 24 schedule. The specific change is to add the following as subparagraph 1 under paragraph b:

> *<u>One-time Exception</u>: Setpoint testing of safety valve MS-RV-70ARV, as required by ASME OM Code Mandatory Appendix I, paragraph I-1320, may be delayed until start of Cycle 24 refueling outage, but no later than June 8, 2008 (90 days from expiration of the 5-year interval on March 10, 2008).*

There are no TS Bases for TS Section 5.5.6.

3.0 Background

The Nuclear System Pressure Relief System consists of eight relief valves (also referred to as safety/relief valves; [SRV]), and three SVs. These valves are located on the main steam lines within the drywell, between the reactor pressure vessel (RPV) and the first main steam isolation valve. The SRVs discharge to the suppression pool through piping connected to the valve. The SRVs provide the following three functions:

1. Overpressure relief operation. By automatic opening, the SRVs limit the pressure rise in the RPV and prevent opening of the SVs.
2. Overpressure safety operation. By automatic opening, the SRVs augment the SVs by opening to prevent nuclear system overpressurization.
3. Depressurization operation. The SRVs are opened automatically or manually as part of the Emergency Core Cooling System (ECCS) function.

The SVs open automatically on pressure to protect against overpressure of the nuclear system. The SVs discharge directly to the interior of the drywell.

The safety objective of the pressure relief system is to prevent over-pressurization of the nuclear system, thereby protecting the reactor coolant pressure boundary from failure, and helping to prevent uncontrolled release of fission products. The automatic depressurization feature works in conjunction with the ECCS to re-flood the core, thereby protecting the nuclear fuel from failure due to overheating.

The American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Mandatory Appendix I, paragraph I-1320, requires that Class 1 pressure relief valves shall be tested at least once every five years. However, it was discovered that a SV currently installed in the Main Steam (MS) System will exceed the five-year test frequency outlined in OM Code, Mandatory Appendix I, paragraph I-1320.

The required extension is related to setpoint testing required by SR 3.4.3.1. This SR requires verifying the safety function lift setpoints of the SRVs and SVs. The frequency of SR 3.4.3.1 is specified as "In accordance with the IST program." The purpose of the SR is to ensure that the subject valves will open at the pressures assumed in the safety analysis contained in the SRV Setpoint Tolerance Analysis for CNS, October 1998. Demonstrating the SRV and SV safety function lift settings requires removing the valves from the plant and shipping them to an offsite test facility. As a result the plant must be in cold shutdown to perform this testing.

The required setpoint testing of this valve had been considered for performance during RFO-23 in fall of 2006. When the scope of work for RFO-23 was being determined, RFO-24 was unofficially scheduled to start in March 2008. Based on that planned outage start date this testing of the valve was postponed until RFO-24.

This condition of MS-RV-70ARV exceeding the five-year frequency for setpoint testing has been entered into the CNS Corrective Action Program.

The above information explains how this condition occurred and why the proposed TS change is necessary.

4.0 Technical Analysis

TS Section 3.4.3, "Safety/Relief Valves (SRVs) and Safety Valves (SVs)," requires the SRVs and SVs to be operable in Reactor Modes 1, 2, and 3. Because setpoint testing of SRVs and SVs is conducted by bench testing of the valve, the SRV or SV must be removed from the plant. As a result the setpoint testing can only be conducted when the plant is shutdown.

The interval for SRV/SV testing specified in OM Code Appendix I is five years. Extension of the five-year interval for SRV testing to coincide with a scheduled refueling outage is allowed by NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants. NUREG-1482, Section 3.1.3, Scheduling of Inservice Tests, discusses scheduling

of tests. Table 3.2 in this section specifies the required frequency for IST activities of various terms, up to a maximum term of two years. This section also discusses the maximum extension of 25 percent of the test interval allowed by TS. However, CNS TS 5.5.6.b allows use of the 25 percent extension only for surveillances with intervals of two years or less. This section of NUREG-1482 also states: "However, licensees should not extend the test intervals for safety and relief valves defined in Appendix I to the OM Code, other than to coincide with a refueling outage."

Setpoint testing of MS-RV-70ARV was last conducted on March 10, 2003. Thus, the five-year test frequency for this valve expires on March 10, 2008. The next refueling outage is currently scheduled to start in April 2008. If this outage starts as scheduled the requested exception would involve an extension of approximately one month for the setpoint testing. An extension of one month is an extension of less than two percent of the allowed interval of 60 months (five years). To accommodate delay in the start of the next refueling outage, NPPD is proposing an extension of 90 days. An extension of 90 days (three months) is a five percent extension of the allowed 60-month interval.

The results of setpoint testing over the last ten years were reviewed. This covers the last three performances of the setpoint testing. The following table summarizes the results of the as found setpoint testing performed on MS-RV-70ARV, as well as the as left values of the setpoint, for the most recent three tests. This shows that the tests were within the currently allowed as found tolerance (range) of plus-or-minus three percent around the nameplate test pressure of 1240 psig. Following completion of the as found tests, the valve was refurbished and the setpoint left within plus-or-minus one percent.

Values of pressure in the table are in units of psig. The value in the Deviation column is the deviation from 1240 psig. Note that the as found acceptance criteria was revised in 1998 from plus-or-minus one percent to plus-or-minus three percent.

Summary of MS-RV-70ARV Setpoint Testing

AS FOUND				AS LEFT	
Date	***Acceptable Range***	***Results***	***Deviation***	***Date***	***Setpoint***
April 9, 1997	1227 to 1253 (± 1%)	1st Act.-1217 2nd Act.-1219	-1.85%	April 11, 1997	1240
October 9, 1998	1202.8 to 1277.2 (± 3%)	1st Act.-1252 2nd Act.-1218	+0.97%	October 15, 1998	1242
March 8, 2003	1202.8 to 1277.2 (± 3%)	1st Act.-1226 2nd Act.-1227	-1.13%	March 10, 2003	1245

Although the results of the as found test performed in April 1997 were outside the acceptable range at that time (plus-or-minus one percent) the results are within the acceptable range in use today (plus-or-minus three percent).

Based on the results presented in the above table, the setpoint drift experienced by MS-RV-70ARV in the three most recent surveillances is acceptable. Based on that, there is reasonable expectation that the as found actuation setpoint of MS-RV-70ARV, following the requested extended period of five years and 90 days, would be within the current acceptable range of plus-or-minus three percent.

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

Nebraska Public Power District (NPPD) is requesting a revision to the Facility Operating License No. DPR-46 for Cooper Nuclear Station (CNS). The requested change proposes to add a provision to Technical Specification (TS) Section 5.5.6, Inservice Testing Program, to allow a one-time extension of the interval for performing setpoint testing of one safety valve.

The Nuclear System Pressure Relief System is comprised of eight safety-relief valves (SRVs) and three safety valves (SVs). Setpoint testing of these SRVs and SVs is performed by means of bench testing. This requires that the installed SRV or SV be removed from the Main Steam System. The plant must be shutdown to do this.

The interval for performing setpoint testing on SRVs and SVs is five years. The next refueling outage is currently scheduled to begin in April 2008. The five-year interval for one SV installed in CNS expires on March 10, 2008, approximately one month before the refueling outage is scheduled to start. To accommodate unanticipated delays in the start of the refueling outage, the requested extension is for a maximum of 90 days beyond the five-year interval.

NPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The function of SRVs and SVs is to prevent overpressurization of the reactor coolant system (RCS) during transients and abnormal operation that could cause increases in RCS pressure. They are also used to depressurize the RCS when needed to allow injection of water from the high-volume, low-pressure Emergency Core Cooling System (ECCS) Low Pressure Coolant Injection mode of the Residual Heat Removal System into the reactor pressure vessel (RPV) as part of mitigation of an accident. Actuation or failure to actuate of a SRV or SV is not an initiator of any accident previously evaluated. Thus, this

proposed amendment would not result in a significant increase in the probability of an accident previously evaluated.

A range or tolerance of plus-or-minus three percent of the setpoint pressure is acceptable for the results of setpoint testing. A 90-day extension of the interval for setpoint testing of one SV is not expected to result in actuation of the SV outside of its acceptable setpoint range. However, even if the single SV whose test interval is being extended did actuate outside of its acceptable range, it is not expected that this would result in a significant degradation in the ability of the Nuclear System Pressure Relief System to perform its safety function, since the remaining eight SRVs and two other SVs would be unaffected by the proposed extension of the testing interval for the single SV. The proposed change does not modify the design of or alter the operation of systems or components used in mitigating design basis accidents. Thus, this proposed amendment would not result in a significant increase in the consequences of any accident previously evaluated.

Based on the above, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

A new or different kind of accident from any previously evaluated might result from a modification of the plant design by either addition of a new system or removal of an existing system, or a change in how any of the plant systems function during the operation of the plant. The proposed change does not modify the plant design, nor does it alter the operation of the plant or equipment involved in either routine plant operation or in the mitigation of the design basis accidents.

Based on the above, it is concluded that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The margin of safety applicable to this issue would be the margin between the pressure at which the SRVs and SVs would actuate and the allowable ASME Code overpressure limit of 1,375 psig (110 percent of vessel design pressure,

1250 psig). This margin would be impacted if the setpoint at which the applicable SV actuated experienced drift greater than the allowable plus-or-minus three percent of the setpoint pressure. This is not expected to occur based on the results demonstrated by the setpoint testing conducted over the last ten years. Those results were two actuations of the SV at a pressure below the nameplate rating with less than two percent deviation, and one actuation at a pressure above the nameplate rating with less than one percent deviation. However, even if this one SV did experience setpoint drift greater than the allowable plus-or-minus three percent, there would not be a significant reduction in the margin since it is expected that the remaining eight SRVs and the two other SVs would actuate within the allowable setpoint tolerance and begin to reduce RCS pressure as needed. Furthermore, the proposed extension will not result in a change to the steam discharge capacity and characteristics of the applicable SV.

Based on the above, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

A. ASME Boiler and Pressure Vessel (B&PV) Code Section III requires that the RPV be protected from overpressure during upset conditions by self-actuated SVs.

CNS complies with this ASME code requirement through the eight SRVs and three SVs in the Nuclear System Pressure Relief System. The requested extension does not significantly challenge in any manner the continued compliance with this code requirement.

B. ASME OM Code, Mandatory Appendix I, Paragraph I-1320, "Test Frequencies, Class 1 Pressure Relief Valves," subparagraph (a) "5-Year Test Interval," requires that Class 1 pressure relief valves be tested at least once every five years. It states that the test interval for any individual valve shall not exceed five years.

The requested one-time extension of testing of the single SV involves an exception to this code requirement. Only the single SV is affected. The other two SVs and the eight SRVs in the CNS Nuclear System Pressure Relief System, continue to comply with this code requirement.

Conclusion

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. NPPD has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS. Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, and sufficient safety margins will be maintained.

In conclusion, based on the considerations discussed above, (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.

6.0 Environmental Consideration

10 CFR 51.22(c) provides categories of actions which are categorical exclusions from performing an environmental assessment. An action which is a categorical exclusion does not require an environmental assessment or an environmental impact statement. 10 CFR 51.22(c)(9) allows as a categorical exclusion issuance of an amendment to a license for a reactor pursuant to 10 CFR Part 50 which changes a SR provided that (1) the amendment involves no significant hazards consideration, (2) there is no significant change in the types or significant increase in the amounts of any effluents that may be released off-site, and (3) there is no significant increase in individual or cumulative occupational radiation exposure.

NPPD has reviewed the proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). The basis for this determination is as follows:

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

2. The proposed license amendment does not introduce any new equipment, nor does it require any existing equipment or systems to perform a different type of function than they are presently designed to perform. NPPD has concluded that this proposed change does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site.

3. These changes do not adversely affect plant systems or operation and therefore, do not significantly increase individual or cumulative occupational radiation exposure beyond that already associated with normal operation.

Therefore, pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the proposed license changes.

7.0 References

7.1 ASME Boiler and Pressure Vessel Code Section III; Article 9, "Protection Against Overpressure"

7.2 NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants"

ATTACHMENT 2

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION FOR A ONE-TIME EXTENSION OF FIVE-YEAR TEST FREQUENCY FOR MS-RV-70ARV

COOPER NUCLEAR STATION
DOCKET NO. 50-298, DPR-46

Technical Specification Page – Markup Format

5.0-10

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves:

a. Testing Frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

1. One-time Exception: Setpoint testing of safety valve MS-RV-70ARV, as required by ASME OM Code Mandatory Appendix I, paragraph I-1320, may be delayed until start of Cycle 24 refueling outage, but no later than June 8, 2008 (90 days from expiration of the 5-year interval on March 10, 2008).

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall be performed once per 18 months for standby service or after 720 hours of

ATTACHMENT 3

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION FOR A ONE-TIME EXTENSION OF FIVE-YEAR TEST FREQUENCY FOR MS-RV-70ARV

COOPER NUCLEAR STATION
DOCKET NO. 50-298, DPR-46

Technical Specification Page – Final Typed Format

5.0-10

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves:

a. Testing Frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

1. One-time Exception: Setpoint testing of safety valve MS-RV-70ARV, as required by ASME OM Code Mandatory Appendix I, paragraph I-1320, may be delayed until start of Cycle 24 refueling outage, but no later than June 8, 2008 (90 days from expiration of the 5-year interval on March 10, 2008).

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall

0.ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2007048

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		