

Mr. J. H. Swailes  
 Vice President of Nuclear Energy  
 Nebraska Public Power District  
 P. O. Box 98  
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March 3, 2000

Template = URR-058

**SUBJECT: COOPER NUCLEAR STATION - ISSUANCE OF AMENDMENT  
 RE: CHANGES TO THE TECHNICAL SPECIFICATIONS (TSs) TO  
 IMPLEMENT 10 CFR PART 50, APPENDIX J, OPTION B, AND CHANGES  
 TO THE TS ASSOCIATED WITH THE CONTAINMENT AIR LOCK INTERLOCK  
 MECHANISM, ISOLATION VALVE TIME TESTING, AND CREDIT FOR  
 ADMINISTRATIVE MEANS FOR SECURING ISOLATION DEVICES  
 (TAC NO. MA6877)**

Dear Mr. Swailes:

The Commission has issued the enclosed Amendment No. 180 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 October 6, 1999, as supplemented by letter dated February 9, 2000.

The amendment addresses the following changes to the TS: (1) provisions for implementation of 10 CFR Part 50, Appendix J, Option B (Technical Specification Task Force (TSTF) Change 52, Revision 2), (2) extension of the required surveillance interval for the containment air lock interlock mechanism from 18 to 24 months (TSTF Change 17, Revision 1), (3) clarification of the valve types requiring isolation time testing (TSTF Change 46, Revision 1), and (4) provisions for use of administrative means for verification of isolation devices that are locked, sealed, or otherwise secured (TSTF Change 269, Revision 2).

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/

Lawrence J. Burkhart, Project Manager, Section 1  
 Project Directorate IV & Decommissioning  
 Division of Licensing Project Management  
 Office of Nuclear Reactor Regulation

Docket No. 50-298

- Enclosures: 1. Amendment No. 180 to DPR-46  
 2. Safety Evaluation

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Cooper Nuclear Station

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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. 180  
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 October 6, 1999, as supplemented by letter dated February 9, 2000, 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.

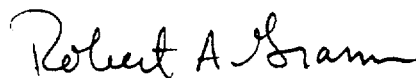
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment: Changes to the Technical  
Specifications

Date of Issuance: March 3, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 180

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 areas of change.

REMOVE

INSERT

1.1-3  
3.6-2  
3.6-7  
3.6-9  
3.6-10  
3.6-13  
3.6-14  
3.6-35  
3.6-37  
5.0-16  
5.0-17  
5.0-18  
5.0-19  
5.0-20  
5.0-21  
-

1.1-3  
3.6-2  
3.6-7  
3.6-9  
3.6-10  
3.6-13  
3.6-14  
3.6-35  
3.6-37  
5.0-16  
5.0-17  
5.0-18  
5.0-19  
5.0-20  
5.0-21  
5.0-22

1.1 Definitions

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DOSE EQUIVALENT I-131  
(continued)

I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 concentration is calculated as follows:  
$$\text{DOSE EQUIVALENT I-131} = (I-131) + 0.0096 (I-132) + 0.18 (I-133) + 0.0025 (I-134) + 0.037 (I-135).$$

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit.

(continued)

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1.1      Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.1.2      Verify drywell to suppression chamber bypass leakage is equivalent to a hole &lt; 1.0 inch in diameter.</p>	<p>18 months</p> <p><u>AND</u></p> <p>————NOTE———— Only required after two consecutive tests fail and continues until two consecutive tests pass</p> <hr/> <p>9 months</p>

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1</p> <hr/> <p style="text-align: center;"><b>NOTES</b></p> <ol style="list-style-type: none"> <li>1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.</li> <li>2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1.</li> </ol> <hr/> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.2.2</p> <p>Verify only one door in the primary containment air lock can be opened at a time.</p>	<p>24 months</p>



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.2</p> <p style="text-align: center;"><u>NOTES</u></p> <p>1. Isolation devices in high radiation areas may be verified by use of administrative means.</p> <p>2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.</p> <hr/> <p>Verify the affected penetration flow path is isolated.</p>	<p>Once per 31 days for isolation devices outside primary containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. ----- One or more penetration flow paths with two PCIVs inoperable except for MSIV leakage not within limit.</p>	<p>B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	<p>1 hour</p>
<p>C. -----NOTE----- Only applicable to penetration flow paths with only one PCIV. ----- One or more penetration flow paths with one PCIV inoperable.</p>	<p>C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p> <p><u>AND</u></p> <p>C.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means.  2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ----- Verify the affected penetration flow path is isolated.</p>	<p>4 hours except for excess flow check valves (EFCVs)</p> <p><u>AND</u></p> <p>12 hours for EFCVs</p> <p>Once per 31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.3.3</p> <p style="text-align: center;"><u>NOTES</u></p> <ol style="list-style-type: none"> <li>1. Valves and blind flanges in high radiation areas may be verified by use of administrative means.</li> <li>2. Not required to be met for PCIVs that are open under administrative controls.</li> </ol> <hr/> <p>Verify each primary containment manual isolation valve and blind flange that is located inside primary containment and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.</p>	<p>Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days</p>
<p>SR 3.6.1.3.4</p> <p>Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.</p>	<p>31 days</p>
<p>SR 3.6.1.3.5</p> <p>Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.</p>	<p>In accordance with the Inservice Testing Program</p>

(continued)

**SURVEILLANCE REQUIREMENTS (continued)**

<b>SURVEILLANCE</b>		<b>FREQUENCY</b>
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq 3$ seconds and $\leq 5$ seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is $\leq 11.5$ scfh when tested at $\geq 29$ psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.2 -----NOTES-----            1. Isolation devices in high radiation areas may be verified by use of administrative means.             2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.            -----             Verify the affected penetration flow path is isolated.</p>	<p>Once per 31 days</p>
<p>B. -----NOTE-----            Only applicable to penetration flow paths with two isolation valves.            -----             One or more penetration flow paths with two SCIVs inoperable.</p>	<p>B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	<p>4 hours</p>
<p>C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.</p>	<p>C.1 Be in MODE 3.   <u>AND</u>            C.2 Be in MODE 4.</p>	<p>12 hours             36 hours</p>

(continued)

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.6.4.2.1</p> <hr/> <p style="text-align: center;"><b>NOTES</b></p> <ol style="list-style-type: none"> <li>1. Valves and blind flanges in high radiation areas may be verified by use of administrative means.</li> <li>2. Not required to be met for SCIVs that are open under administrative controls.</li> </ol> <hr/> <p>Verify each secondary containment isolation manual valve and blind flange that is required to be closed during accident conditions is closed.</p>	<p>31 days</p>
<p>SR 3.6.4.2.2</p> <p>Verify the isolation time of each power operated automatic SCIV is within limits.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.6.4.2.3</p> <p>Verify each automatic SCIV actuates to the isolation position on an actual or simulated actuation signal.</p>	<p>18 months</p>

## 5.5 Programs and Manuals

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### 5.5.11 Safety Function Determination Program (SFDP) (continued)

For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
3. A required system redundant to support system(s) for the supported systems b.1 and b.2 above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

### 5.5.12 Primary Containment Leakage Rate Testing Program

- a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:
  1. Exemption from Appendix J to 10CFR Part 50 to allow reverse direction local leak rate testing of four containment isolation valves at Cooper Nuclear Station (TAC NO. M89769) (July 22, 1994).
  2. Exemption from Appendix J to 10CFR Part 50 to allow MSIV testing at 29 psig and expansion bellows testing at 5 psig between the plies (Sept. 16, 1977).
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 58.0 psig. The containment design pressure is 56.0 psig.
- c. The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.635% of containment air weight per day.

## 5.5 Programs and Manuals

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### 5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- d. Leakage Rate acceptance criteria are:
1. Containment leakage rate acceptance criterion is  $\leq 1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are,  $<0.60 L_a$  for the Type B and C tests and  $\leq 0.75 L_a$  for Type A tests.
  2. Air lock testing acceptance criteria are:
    - a. Overall air lock leakage rate is  $\leq 12$  scfh when tested at  $\geq P_a$ .
    - b. Overall air lock leakage rate is  $\leq 0.23$  scfh when tested at  $\geq 3.0$  psig.
- e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.
- f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Reporting Requirements

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The following reports shall be submitted in accordance with 10 CFR 50.4.

#### 5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures  $> 100$  mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on electronic or pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling  $< 20\%$  of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

#### 5.6.2 Annual Radiological Environmental Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Assessment Manual (ODAM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODA, as well as summarized and tabulated results of these analyses and measurements in the format of the table in Regulatory Guide 4.8, December 1975. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

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5.6 Reporting Requirements (continued)

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**5.6.3 Radioactive Effluent Release Report**

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODAM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

**5.6.4 Monthly Operating Reports**

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

**5.6.5 CORE OPERATING LIMITS REPORT (COLR)**

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
  1. The Average Planar Linear Heat Generation Rates for Specification 3.2.1.
  2. The Minimum Critical Power Ratio for Specifications 3.2.2 and 3.7.7.
  3. The three Rod Block Monitor Upscale Allowable Values for Specification 3.3.2.1.
  4. The power/flow map defining the Stability Exclusion Region for Specification 3.4.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
  1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (Revision specified in the COLR).

(continued)

5.6 Reporting Requirements

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. NEDE-23785-1-P-A, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident", Volume III, Revision 1, October 1984.
  3. NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology" (the approved Revision at the time the reload analysis is performed).
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

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- 5.7.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the deep dose equivalent in excess of 100 mrem but less than 1000 mrem in one hour (measurement made at 12 inches from source of radiation) shall be barricaded (barricade will impede physical movement across the entrance or access to the high radiation area; i.e., doors, yellow and magenta rope, turnstile) and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP). Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the SWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:
- a. A monitoring device which continuously indicates the radiation dose rate in the area.
  - b. A monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.
  - c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures), with a dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic dose rate monitoring at the frequency specified by Health Physics supervision.
- 5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with dose rates such that a major portion of the body could receive in 1 hour a deep dose equivalent in excess of 1000 mrem (measurement made at 12 inches from source of radiation) shall be provided with locked doors to prevent unauthorized entry. Doors shall remain locked except during periods of access by personnel under an approved SWP which shall specify the dose rates in the immediate work area. For individual high radiation areas accessible to personnel that are located within large areas, such as the containment, or areas where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, then that area shall be barricaded and conspicuously posted. Area radiation monitors that have been set to alarm if radiation levels increase,

(continued)

5.7 High Radiation Area

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5.7.2 (continued)

provide both a visual and an audible signal to alert personnel in the area of the increase. These monitors may be used to meet Specification 5.7.1.a provided that the dose rates and alarms have been established by radiation protection personnel. Stay times or continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide additional positive exposure control over the activities within the area.

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

## 1.0 INTRODUCTION

By application dated October 6, 1999, as supplemented by letter dated February 9, 2000, Nebraska Public Power District (NPPD, the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (Cooper). The proposed changes would (1) provide for implementation of 10 CFR Part 50, Appendix J, Option B (Technical Specification Task Force (TSTF) Change 52, Revision 2), (2) extend the required surveillance interval for the containment air lock interlock mechanism from 18 to 24 months (TSTF Change 17, Revision 1), (3) clarify the valve types requiring isolation time testing (TSTF Change 46, Revision 1), and (4) provide for use of administrative means for verification of isolation devices that are locked, sealed, or otherwise secured (TSTF Change 269, Revision 2). The February 9, 2000, supplement provided clarifying information that was within the scope of the October 6, 1999, application and the staff's original *Federal Register* notice and did not change the staff's initial proposed no significant hazards consideration determination.

## 2.0 BACKGROUND

### 2.1 Implementation of 10 CFR Part 50, Appendix J, Option B

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the *Federal Register* (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety that impose a significant regulatory burden. The requirements of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," were considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Containment Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the staff approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the *Federal Register* on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance, following the staff approval of the licensee's plant-specific application.

Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, was developed as a method acceptable to the staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the staff for complying with Option B with four exceptions which are described therein.

Option B requires that RG 1.163 or another implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the proposed Cooper TS.

The guidance contained in RG 1.163 specifies an extension in Type A test (primary containment integrated leak-rate test) frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests (mostly associated with leak-rate testing of components with resilient seals) may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests. Type C tests (containment isolation valve leak-rate tests) may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed generic TS to implement Option B. After some discussion, the staff and NEI agreed on final generic TS which were transmitted to NEI in a letter dated November 2, 1995. These generic TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of, or affect, performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to staff inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

By application dated October 6, 1999, as supplemented by letter dated February 9, 2000, the licensee requested changes to the Cooper TS in order to implement Option B for Type A, B, and C tests. The proposed TS changes, in general, remove details of the Appendix J test program, from the Definition and Surveillance Requirements sections of the TSs, and

incorporate these details in a new, proposed TS in the Section 5.0, "Administrative Controls," of the TSs.

## 2.2 Surveillance for Containment Air Lock Interlock Mechanism

At the present time, the containment air lock interlock mechanism is required to undergo surveillance, per Surveillance Requirement (SR) 3.6.1.2.2, to demonstrate the operability of the interlock mechanism, using an 18-month frequency. The subject surveillance requires the licensee to "Verify only one door in the primary containment air lock can be opened at a time." The licensee has proposed to change the surveillance frequency from 18 to 24 months.

## 2.3 Valve Types Requiring Isolation Time Testing

At the present time SR 3.6.1.3.5 and SR 3.6.4.2.2 require that certain primary containment isolation valves (PCIVs) and secondary containment isolation valves (SCIVs), respectively, undergo isolation timing every 31 days. The SR is as follows, "Verify the isolation time of each power operated and each automatic [PCIV or SCIV]...is within limits." The licensee has proposed the deletion of reference to "power operated" valves and substitute a reference to "power operated, automatic" valves.

## 2.4 Administrative Means For Verification of Secured Devices

At the present time, TS 3.6.1.3 requires that the licensee "Verify the affected penetration flow path is isolated" under Condition A., "One or more penetration flow paths with one PCIV inoperable except for MSIV [main steam isolation valve] leakage not within limit" and Condition C., "One or more penetration flow paths with one PCIV inoperable". In addition, TS 3.6.4.2 requires that the licensee "Verify the affected penetration flow path is isolated" under Condition A., "One or more penetration flow paths with one SCIV inoperable". A note in the corresponding Required Actions, in the TS 3.6.1.3 (A.2 and C.2) and TS 3.6.4.2 (A.2), allows that, "Isolation devices in high radiation areas may be verified by use of administrative means." The licensee has proposed a second note to allow that, "Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means".

## 3.0 EVALUATION

### 3.1 Implementation of 10 CFR Part 50, Appendix J, Option B

The licensee's application dated October 6, 1999, as supplemented by letter dated February 9, 2000, proposes to establish a "Primary Containment Leakage Rate Testing Program" and proposes to add this program as TS 5.5.12. The program references RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, which specifies methods acceptable to the staff for complying with Option B of Appendix J to 10 CFR Part 50. Option B permits a licensee to choose Type A, or Type B and C, or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The licensee has proposed that information concerning maximum primary containment leakage ( $L_a$ ), at the peak containment pressure ( $P_a$ ) be deleted from the TS 1.1 definition for "Dose



Equivalent I-131." This information is contained in the proposed TS 5.5.12 and is, therefore, acceptable.

The licensee has proposed to remove information concerning conduct of primary containment air lock testing and its leakage acceptance criteria from SR 3.6.1.1.1 and SR 3.6.1.2.1. Conduct of the testing would no longer be performed in accordance with "10 CFR 50, Appendix J, Option A, as modified by approved exemptions," but would be performed in accordance with "the Primary Containment Leakage Rate testing Program." These changes are acceptable because 10 CFR Part 50 Appendix J, Option A, is no longer applicable due to NPPD adopting 10 CFR Part 50 Appendix J, Option B, and the leakage acceptance criteria are stated explicitly in TS 5.5.12.

The guidance governing frequency of SR 3.6.1.1.1 and SR 3.6.1.2.1 for primary containment air lock testing, and SR 3.6.1.3.10 for verification of leakage rate through each MSIV will be changed from "10 CFR 50 Appendix J, Option A, as modified by approved exemptions" to "the Primary Containment Leakage Rate Testing Program" as required by TS 5.5.12. This change is acceptable because 10 CFR Part 50 Appendix J, Option A, is no longer applicable due to NPPD adopting 10 CFR Part 50 Appendix J, Option B. Option B of 10 CFR Part 50 Appendix J allows adoption of a performance-based testing program that will meet the requirements of proposed TS 5.5.12. Therefore, this change is acceptable.

NPPD proposes to delete the Note that SR 3.0.2 is not applicable with respect to the frequency of SR 3.6.1.1.1 and SR 3.6.1.2.1 for primary containment air lock testing, and SR 3.6.1.3.10 for verification of leakage rate through each MSIV. The frequency of each of the SRs will require a frequency in accordance with the Primary Containment Leakage Rate Testing Program, which must meet the requirements of proposed TS 5.5.12. TS 5.5.12 explicitly states that the provisions of SR 3.0.2 do not apply to this program. Therefore, the change is acceptable.

The licensee has proposed a new TS 5.5.12, "Primary Containment Leakage Rate Testing Program." The features of this new TS would be as follows:

- Section "a." would establish the containment test program via reference to 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. Additionally, the program would be in accordance with RG 1.163 as modified by two exceptions. The exceptions would be referenced in proposed Section "a.1" for the exemptions issued on July 22, 1994, to allow reverse testing of four containment isolation valves and proposed Section "a.2" for the exemption to allow MSIV testing at 29 psig and expansion bellows testing of 5 psig between the plies. In this regard, the licensee's October 6, 1999, application requests that the staff withdraw two additional exemptions, issued on November 30, 1995, and September 3, 1982, regarding Appendix J for Cooper. The staff acknowledges that the two additional Appendix J exemptions, referenced in the licensee's October 6, 1999, application, apply to Option A of Appendix J but not to Option B and are, therefore, no longer needed by the licensee.
- Section "b." would provide a value of 58.0 psig for " $P_a$ " and reference the containment design pressure of 56.0 psig.

- Section "c." would provide a value of 0.635% for " $L_a$ " at the stated value of " $P_a$ ". This defines " $L_a$ " as it was previously defined in TS 1.1.
- Section "d." would provide the leakage rate acceptance criteria as previously contained in SR 3.6.1.1.1 and SR 3.6.1.2.1 for primary containment air lock testing.
- Section "e." would state that the provisions of SR 3.0.2 regarding test frequencies do not apply to the Primary Containment Testing Program. This was previously noted in the frequency column of SR 3.6.1.1.1 and SR 3.6.1.2.1 for primary containment air lock testing and SR 3.6.1.3.10 for verification of leakage rate through each MSIV.
- Section "f." would state that the provisions of SR 3.0.3, regarding remedial action to be taken when a Limiting Condition for Operation, and the associated Action, are not met, do apply to the Primary Containment Testing Program.

The staff has reviewed the changes to the TSs and associated Bases proposed by the licensee, for Option B implementation. The staff concludes that the proposed TS Section 5.5.12, and accordingly all of the changes described above, meet the regulatory requirements for implementing a performance-based primary containment leakage rate testing program as allowed by adoption of 10 CFR Part 50, Appendix J, Option B. Therefore, the changes described above are acceptable.

### 3.2 Surveillance for Containment Air Lock Interlock Mechanism

As noted in Section 2.2, herein, the licensee has proposed revising the surveillance frequency for the required testing of the air lock interlock mechanism, in SR 3.6.1.2.2, from 18 to 24 months. Typically, the interlock mechanism is activated after each refueling outage, verified operable via SR 3.6.1.2.2, and not disturbed until the next refueling outage. If the need for maintenance arises when the interlock mechanism is required to be operable, the performance of SR 3.6.1.2.2 would be required following the maintenance. In addition, when an air lock is opened when the interlock mechanism is required to be operable, the operator first verifies that one door is completely shut and the door seals pressurized before attempting to open the other door; therefore, the interlock mechanism is not challenged except during the actual testing of the interlock mechanism. Consequently, testing the interlock on a 24-month interval would be sufficient to ensure proper operation of the interlock mechanism.

Testing of the air lock interlock mechanism is accomplished through having one door not completely engaged in the closed position, while attempting to open the second door. Failure of this surveillance effectively results in a loss of containment integrity. Procedures and training do not allow this interlock mechanism to be challenged for normal ingress and egress. For normal ingress and egress, one door is opened, all personnel and equipment as necessary are placed into the air lock, and then the door is completely closed prior to attempting to open the second door. The performance of SR 3.6.1.2.2 is contrary to processes and training of conservative operation when operability of the interlock function is required. Testing with the plant in a condition where the interlock is not required to be operable (e.g., during a refueling outage, a 24-month interval) is desirable. The failure rate of this physical device is low based on the design of the interlock mechanism.

Historically, the air lock interlock mechanism surveillance has had its frequency chosen to coincide with the frequency of the overall air lock leakage test. According to 10 CFR Part 50, Appendix J, Option A, this frequency is once per 6 months. However, Appendix J, Option B, allows for an extension of the overall air lock leakage test frequency to a maximum of 30 months.

Based upon the above, the staff concludes that the licensee's proposal to change the required frequency for SR 3.6.1.2.2, to 24 months and the associated Bases are acceptable.

### 3.3 Valve Types Requiring Isolation Time Testing

The Bases for SR 3.6.1.3.5 and 3.6.4.2.2 state, "The isolation time test ensures that the [SCIV or valve] will isolate in a time period less than or equal to that assumed in the safety analyses." There may be valves credited as containment isolation valves that are power-operated (i.e., can be remotely operated) that do not receive a containment isolation signal. These power-operated valves do not have an isolation time as assumed in the accident analyses since they require operator action; therefore, deleting the reference to the power isolation valve time testing reduces the potential for misinterpreting the requirements of SR 3.6.1.3.5 and 3.6.4.2.2 while maintaining the assumptions of the accident analysis. Accordingly, the staff finds the proposed changes to SR 3.6.1.3.5, SR 3.6.4.2.2 and associated Bases to be acceptable.

### 3.4 Administrative Means for Verification of Secured Devices

With regard to crediting administrative means for verification of secured devices, it is reasonable to assume that the initial establishment of component status (e.g., a closure of an isolation valve) was performed correctly. Subsequent verification is intended to ensure that the component has not been inadvertently repositioned. Given that the function of locking, sealing, or otherwise securing components is to ensure the same avoidance of inadvertent repositioning, the periodic reverification should be only a verification of the administrative controls that ensure that the component remains in the required state. It would be inappropriate to remove the lock, seal, or other means of securing the component solely to perform an active verification of the required state. Accordingly, the staff concludes that the proposed changes to TS 3.6.1.3, Required Action A.2 and Required Action C.2, and TS 3.6.4.2, Required Action A.2, and associated Bases are acceptable.

## 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 comment.

## 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 surveillance requirements. The 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 (64 FR 73092). 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.

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