January 7, 1993

Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M83785)

The Commission has issued the enclosed Amendment No.157 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated May 4, 1992.

The amendment changes the Technical Specifications by removing Table 3.7.2, "Testable Penetrations with Double O-Ring Seals," Table 3.7.3, "Testable Penetrations with Testable Bellows," and Table 3.7.4 "Primary Containment Testable Isolation Valves." The lists removed from the TS will be incorporated into plant procedures and the Updated Safety Analysis Report, which are subject to the administrative control of TS 6.2.1.A.4. Guidance on the proposed TS changes was provided by NRC Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Harry Rood, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 157 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	NRC/Local PDR	PD4-1 Reading	H. Rood(2)
M. Virgilio	J. Larkins	P. Noonan	J. Roe
ACRS(10)(MSP315)	OGC(MS15B18)	D. Hagan(MS3206)	G.Hill (4)
Wanda Jones(MS7103)	C. Grimes(MS11E22)	PD4-1 Plant File	OPA(MS2G5)
E. Collins, RIV	OC/LFMB(MS4503)	J. Gagliardo, RIV	C. Yates

OFC	LA:PD4-1	I:PD4-1	PM:PD4-1	OGC	D:PD4-1
NAME	PNoonan	CYates:pk	HRood	EHollen	JLarkins
DATE	12/7/92	12/7/92	12/8/92	12/10/92	1/7/93

OFFICIAL RECORD COPY Document Name: C0083785.amd

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

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Sincerely,

Harry Rood

Harry Rood, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
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Enclosures:
1. Amendment No. 157 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Guy R. Horn
Nuclear Power Group Manager

Cooper Nuclear Station

cc:

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
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 May 4, 1992, 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. 157 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W O'Connor

for John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 7, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

REMOVE PAGES	INSERT PAGES
162	162
162a	162a
167	167
168	168
169	169
171	171
172	-
173	-
174	-
175	-
178	178
183	183

LIMITING CONDITIONS FOR OPERATION

3.7.A (Cont'd)

SURVEILLANCE REQUIREMENTS

4.7.A.2.e (cont'd)

repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. Local Leak Rate Tests

1. With the exceptions specified below, local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. The test duration of all valves and penetrations shall be of sufficient length to determine repeatable results. The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La.

2. Bolted double-gasket seals shall be tested after each opening and during each reactor shutdown for refueling, or other convenient intervals but in no case at intervals greater than two years.

3. The main steam isolation valves (MSIV's) shall be tested at a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repairs and retest shall be performed to correct the condition. This is an exemption to Appendix J of 10CFR50.

3.7.A (Cont'd)

4.7.A.2.f (cont'd)

4. Main steam line and feedwater line expansion bellows shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50.

5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50.

   The maximum allowable leakage at a test pressure of 58 psig is 12 scfh. Leakage measured at test pressure less than 58 psig is adjusted to the equivalent value at 58 psig.

g. Deleted

h. Drywell Surfaces

   The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage.

3.7.D (cont'd.)

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolated condition.*

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

4.7.D (cont'd.)

b. At least once per quarter:

(1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 75%, trip main steam isolation valves individually and verify closure time.

c. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

d. At least once per operating cycle, while shutdown, the devices that limit the maximum opening angle to 60° shall be verified functional for the following valves: PC-230MV, PC-231MV, PC-232MV, and PC-233MV.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 1)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
Main Steam Isolation Valves					
MS-AO-80- A,B,C, & D	4		3 § T § 5	O	GC
MS-AO-86- A,B,C, & D		4	3 § T § 5	O	GC
Drywell Floor Drain Iso. Valves RW-AO-82, RW-AO-83		2	15	O	GC
Drywell Equipment Drain Iso. Valves RW-AO-94, RW-AO-95		2	15	O	GC
Main Steam Line Drain Valves MS-MO-74, MS-MO-77	1	1	30	O	GC
Reactor Water Sample Valves RR-740AV, RR-741AV	1	1	15	O	GC
Reactor Water Cleanup System Iso. Valves RWCU-MO-15, RWCU-MO-18	1	1	60	O	GC
RHR Suction Cooling Iso. Valve RHR-MO-17, RHR-MO-18	1	1	40	C	SC
RHR Discharge to Radwaste Iso. Valves RHR-MO-57, RHR-MO-67		2	20	C	SC
Suppression Chamber Purge & Vent PC-245AV, PC-230MV		2	15	C	SC
Suppression Chamber N_2 Supply PC-237AV, PC-233MV		2	15	C	SC

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 2)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
Primary Containment Purge & Vent PC-246AV, PC-231MV		2	15	C	SC
Primary Containment & N_2 Supply PC-238AV, PC-232MV		2	15	C	SC
Suppression Chamber Purge & Vent PC-230MV Bypass (PC-305MV)		1	40	C	SC(4)
Primary Containment Purge & Vent PC-231MV Bypass (PC-306MV)		1	40	C	SC(4)
Dilution Supply					
PC-1303MV, PC-1304MV		2	15	C	SC
PC-1305MV, PC-1306MV		2	15	C	SC
Dilution Supply					
PC-1301MV, PC-1302MV		2	15	O	GC
PC-1311MV, PC-1312MV		2	15	O	GC
Suppression Chamber Purge and Vent Exhaust PC-1308MV		1	15	C	SC
Primary Containment Purge and Vent Exhaust PC-1310MV		1	15	C	SC

Pages 171, 172, 173, 174, and 175 have been deleted.

trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Certain isolation valves are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore, the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows.

Surveillance requirements for integrity of the personnel air lock are specified in Enclosure 1 (Exemption) to the letter, D. G. Eisenhut to J. M. Pilant, September 3, 1982. When the Personnel Air Lock Leakage Test is performed at a test pressure less than 58 psig, the measured leakage must be adjusted to reflect the expected leakage at 58 psig. Equation A-3 of Enclosure 3 (Franklin Research Center Technical Evaluation Report) to the letter, D. G. Eisenhut to J. M. Pilant, September 3, 1982, defines the method of adjustment.

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily

4.7.B & 4.7.C BASES

Table 5.1 of ANSI N509-1980. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52, Revision 2, March, 1978.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one Standby Gas Treatment subsystem is inoperable, the operable subsystem's operability is verified daily. This substantiates the availability of the operable subsystem and thus reactor operation or refueling operation can continue for a limited period of time.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

The USAR identifies those testable primary containment valves that perform an isolation function, and testable penetrations with Double O-Ring Seals, and testable penetrations with testable Bellows ensuring that any changes thereto receive a 10CFR50.59 review. In addition, plant procedures also identify containment isolation valves, and testable penetrations with Double O-Ring Seals, and testable penetrations with testable Bellows changes to these procedures and the USAR are controlled by Technical Specification 6.2.1.A.4 (Administrative Controls).

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated May 4, 1992, Nebraska Public Power District (the licensee) submitted a request for changes to the Cooper Nuclear Station Technical Specifications (TS). The requested changes would remove the TS tables that include list of components referenced in individual specifications.
In addition, the TS requirements have been modified to remove all references to these tables. Finally, the TS have been modified to state requirements in general terms that include components listed in the tables removed from the TS. Guidance on the proposed TS changes was provided by NRC Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991.

2.0 EVALUATION

The licensee has proposed the removal of Table 3.7.2, "Testable Penetrations with Double O-Ring Seals," and to revise TS 4.7.A.2.f.1 and TS 4.7.A.2.f.2 to remove references to Table 3.7.2.

The licensee has also proposed the removal of Table 3.7.3, "Testable Penetrations with Testable Bellows," as well as to revise TS 4.7.A.2.f.1 and TS 4.7.A.2.f.4 to remove references to Table 3.7.3.

In addition, the licensee has also proposed the removal of Table 3.7.4, "Primary Containment Testable Isolation Valves," and to revise TS 4.7.A.2.f.1 to remove references to Table 3.7.4.

With the removal of the Table of primary containment testable isolation valves, the operability requirements have been stated in general terms that apply to all containment isolation valves including those that are closed or locked closed. These valves are locked or sealed closed consistent with the regulatory requirements for manually-operated valves that are used as containment isolation valves. Because opening these valves would be contrary to the operability requirements of these valves, the following footnote to the Limiting Condition for Operation under TS 3.7.D.2 has been proposed:

* Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

With the removal of Table 3.7.4, the licensee has proposed to add a new surveillance requirement, TS 4.7.D.1.d. This new specification was a footnote to Table 3.7.4, which discusses the surveillance requirements for devices installed to limit the opening angle of certain containment isolation valves. The new surveillance requirement will state as follows:

> At least once per operating cycle while shutdown, the devices that limit the maximum opening angle to 60° shall be verified functional for the following valves: PC-230MV, PC-231MV, PC-232MV, and PC-233MV.

Although GL 91-08 would allow its removal, the licensee has elected to retain Table 3.7.1, "Primary Containment Isolation Valves," in the TS. However, the licensee has proposed corrections to Table 3.7.1. Specifically, corrections are proposed for the component identification code (CIC) numbers for the reactor water sample valves and the CIC numbers for the air containment atmosphere dilution (ACAD) valves to indicate they are listed as primary containment valves. Because Table 3.7.1 will remain, TS 3.7.D will continue to state as follows:

> During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

The licensee has proposed changes to the above TS that are consistent with the guidance provided in GL 91-08. In addition, the licensee has confirmed that component lists removed from the TS have been updated to identify all components to which the TS requirements apply and are located in controlled plant procedures.

On the basis of its review of this matter, the NRC staff finds that the proposed changes to the TS for Cooper Nuclear Station are primarily administrative and do not alter the requirements set forth in the existing TS. Overall, these changes will allow the licensee to make corrections and updates to the lists of components to which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS. Therefore, the staff finds that the proposed TS changes are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a surveillance requirement. 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) (57 FR 30252). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

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

Principal Contributor: C. Yates

Date: January 7, 1993