April 23, 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. 162 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M84744)

The Commission has issued the enclosed Amendment No. 162 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of editorial revisions to the Technical Specifications (TS) in response to your application dated September 9, 1992.

Specifically, the amendment modifies TS Table 3.1.1, Reactor Protection System Instrumentation Requirements, to indicate that the Main Steam Line Isolation Valve Closure trip is only required while the plant is in the RUN mode of operation. Also, the amendment adds bases for the instrument settings for the Core Spray and the Low Pressure Coolant Injection mode of the Residual Heat Removal systems to the TS Bases section for the Core Standby Cooling System. The amendment also changes TS Limiting Condition for Operation (LCO 3.12.A.2.c to clarify that only one emergency bypass fan exists in the Control Room Emergency Filter system. Finally, minor editorial changes and typographical error corrections are made to TS pages iii, 48, 63, 78, 87, 206, 209a, 215, 215a, 215d, 215e, and 215f.

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. 162 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	NRC/Local PDR	PD4-1 Reading
J. Roe	M. Virgilio	JPellet
H. Rood(2)	P. Noonan	PD4-1 Plant File
ACRS(10)(MSP315)	OGC(MS15B18)	J. Gagliardo, RIV
D. Hagan(MS3206)	RIVOPA(MS2G5)	E. Collins, RIV
OC/LFMB(MS4503)	G. Hill (2)	R. Kopriva, RIV
Wanda Jones(MS7103)	C. Grimes (11E22)	

OFC	LA:PD4-1	PM:PD4-1	BC:SCSB	OGC	D(A):PD4-1
NAME	PNoonan	HRood:pk	RJones	MZOBLER	JPellet
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Harry Rood

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Project Directorate IV-1
Division of Reactor Projects - III/IV/V
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Enclosures:
1. Amendment No. 162 to
   License No. DPR-46
2. Safety Evaluation

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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. 162
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 September 9, 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. 162, 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

John L. Pellet, Acting 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: April 23, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 162

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
iii	iii
29	29
48	48
63	78
78	85
85	87
87	206
206	209a
209a	209a
215	215
215a	215a
215d	215d
215e	215e
215f	215f

TABLE OF CONTENTS (cont'd)

COOPER NUCLEAR STATION
TABLE 3.1.1 (Page 2)
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Required When Equipment Operability is Not Assured (1)
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D				X(6)	≤ 10% of valve closure	4 4	A or C A or C
Turbine Control Valve Fast Closure TGF-63/OPC-1,2,3,4				X(4)	≥ 1000 psig turbine control fluid	2	A or B
Turbine Stop Valve Closure SVOS-1(1), SVOS-1(2) SVOS-2(1), SVOS-2(2)				X(4)	≤ 10% of valve Closure	2	A or B
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	≤ 30% first stage press..	2	A or B

LIMITING CONDITIONS FOR OPERATION

3.2 (cont'd.)

D. Radiation Monitoring Systems -Isolation & Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

a. Operability of the Steam Jet Air Ejector Off-Gas System monitor is defined in Table 3.21.A.2.

b. The time delay setting for closure of the steam jet air ejector isolation valves shall not exceed 15 minutes.

c. Other limiting conditions for operation are given on Table 3.2.D and Specifications 3.21.A.2 and 3.21.C.6.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

The limiting conditions for operation are given on Table 3.2.D.

3. Liquid Radwaste Discharge Isolation

The limiting conditions for operation are given on Table 3.2.D and Specification 3.21.B.

4. Control Room Emergency Filter System

The limiting conditions for operation are given on Table 3.2.D and the Section entitled "Additional Safety Related Plant Capabilities."

5. Mechanical Vacuum Pump Isolation

a. The mechanical vacuum pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel whenever the main steam isolation valves are open.

b. If the limits of 3.2.D.5.a are not met, the vacuum pump shall be isolated.

SURVEILLANCE REQUIREMENTS

4.2 (cont'd.)

D. Radiation Monitoring Systems -Isolation & Initiation Functions

1. Steam Jet Air Ejector Off-Gas System

Instrumentation surveillance requirements are given on Table 4.2.D.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

Instrumentation surveillance requirements are given on Table 4.2.D.

3. Liquid Radwaste Discharge Isolation

Instrumentation surveillance requirements are given on Table 4.2.D.

4. Control Room Emergency Filter System

The instrument surveillance requirements are given on Table 4.2.D.

5. Mechanical Vacuum Pump Isolation

The instrument surveillance requirements are given on Tables 4.2.A, and 4.2.D.

Amendment No. ~~80,89,147~~,162

COOPER NUCLEAR STATION

TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I. D. No.	Setting Limit	Number of Sensor Channels Provided by Design	Action (1)
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	(3)	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A, B, C & D	≤ 100 mr/hr	4	B
Liquid Radwaste Discharge Isolation	RMP-RM-1	(2)	1	C
Control Room Emergency Filter	RMV-RM-1	$4x10^3$ CPM	1	D
Mechanical Vacuum Pump Isolation (4)	RMP-RM-251 A, B, C & D	3 times normal full power background. Alarm at 1.5 times normal full power background	4	E

COOPER NUCLEAR STATION
TABLE 4.2.D
MINIMUM TEST AND CALIBRATION FREQUENCIES FOR RADIATION MONITORING SYSTEMS

System	Instrument I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	(12)	(12)	(12)
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A,B,C&D	(12)	(12)	(12)
Liquid Radwaste Discharge Isolation	RMP-RM-1	(11)	(11)	(11)
Control Room Emergency Filter	RMV-RM-1	Once/Month (1)	Once/3 Months	Once/Day
Mechanical Vacuum Pump Isolation	RMP-RM-251 A, B, C & D		See Table 4.2.A	
Logic Systems				
SJAE Off-Gas Isolation		Once/18 Months		
Standby Gas Treatment Initiation		Once/18 Months		
Reactor Building Isolation		Once/18 Months		
Liquid Radwaste Disch. Isolation		Once/6 Months		
Control Room Emergency Filter		Once/6 Months		
Mechanical Vacuum Pump Isolation		Once/Operating Cycle		

3.2 BASES (cont'd)

B. Core and Containment Cooling Systems Initiation and Control

The instrumentation which initiates Core Standby Cooling System (CSCS) action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

CORE SPRAY

Initiation and control instrumentation settings ensure that the Core Spray system operates to ensure fuel cladding temperatures do not exceed 2200°F during a design basis LOCA. The basis for the settings is discussed in USAR Section VII-4.

RESIDUAL HEAT REMOVAL (LPCI MODE)

Initiation and control instrumentation settings ensure that the LPCI mode of the Residual Heat Removal system operates to ensure fuel cladding temperatures do not exceed 2200°F during a design basis LOCA. High drywell pressure and reactor water level instrumentation also allow injection water to be diverted for containment spray. The basis for the settings is discussed in USAR Section VII-4.

HPCI

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping including the RHR Condensing Mode Steam. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

Temperature is monitored at twelve (12) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of ≤ 300% of design flow for high flow and ≤ 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

RCIC

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of ≤ 300% for high flow and ≤ 200°F for temperature are based on the same criteria as the HPCI.

ADS

The effective emergency core cooling for small pipe breaks, the HPCI system, must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

3.2 BASES (Cont'd)

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip setting of 1.0 ci/sec (prior to 30 min. delay) provides an improved capability to detect fuel pin cladding failures to allow prevention of serious degradation of fuel pin cladding integrity which might result from plant operation with a misoriented or misloaded fuel assembly. This limit is more restrictive than 0.39 ci/sec noble gas release rate at the air ejectors (after 30 min. delay) which was used as the source term for an accident analysis of the augmented off-gas system. Using the .39 ci/sec source term, the maximum off-site total body dose would be less than the .5 rem limit.

2. Reactor Building Isolation and Standby Gas Treatment Initiation

Reactor Building Isolation and Standby Gas Treatment initiation is provided in a 1-out-of-2 taken twice logic design via four radiation sensors located on the Reactor Building ventilation exhaust plenum. Each trip system (division) consists of two channels with a 1-out-of-2 logic for upscale trips, and a 2-out-of-2 logic for downscale trips. This trip function is provided to limit the release of radioactivity resulting from a refueling (fuel handling) accident.

Trip settings of <100 mr/hr for the monitors in the ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3. Liquid Radwaste Discharge Isolation

The liquid radwaste monitor assures that all liquid discharged to the discharge canal does not exceed the limits of Specification 3.21.B. Upon sensing a high discharge level, an isolation signal is generated which closes the radwaste discharge valve. The set point is adjustable to compensate for variable isotopic discharges and dilution flow rates.

4. Control Room Emergency Filter System

The main control room ventilation isolation is provided by a detector monitoring the intake of the control room ventilation system. Automatic isolation of the normal supply and exhaust and the activation of the emergency filter system is provided by the radiation detector trip function at the predetermined trip level.

5. Mechanical Vacuum Pump

The mechanical vacuum pump isolation prevents the exhausting of radioactive gas thru the 1 minute holdup line upon receipt of a main steam line high radiation signal.

E. Drywell Leak Detection

Flow transmitters are used to record the flow of liquid from the drywell sumps. An air sampling system is also provided to detect leakage inside the primary containment.

3.10 (Cont'd)

G. Control Room Emergency Filter System

From and after the date that the Control Room Emergency Filter system is made or found to be inoperable for any reason, refueling operations are permissible only during the succeeding seven days unless the system is sooner made operable. If these conditions cannot be met, refueling operations shall be terminated in an orderly manner.

H. Spent Fuel Cask Handling

1. Fuel cask handling above the 931' level of the Reactor Building will be done in the RESTRICTED MODE only except as specified in 3.10.H.2.

2. Fuel cask handling in other than the RESTRICTED MODE will be permitted in emergency or equipment failure situations only to the extent necessary to get the cask to the closest acceptable stable location.

3. Operation with a failed controlled area limit switch is permissible for 48 hours providing an operator is on the refueling floor to assure the crane is operated within the restricted zone painted on the floor.

4. Spent fuel casks weighing in excess of 140,000 lbs. shall not be handled.

4.10 (Cont'd)

H. Spent Fuel Cask Handling

1. Prior to fuel cask handling operations, the redundant crane including the rope, hooks, slings, shackles and other operating mechanisms will be inspected.

   The rope will be replaced if any of the following conditions exist:

   a. Twelve (12) randomly distributed broken wires in one lay or four (4) broken wires in one strand of one rope lay.

   b. Wear of one-third the original diameter of outside individual wire.

   c. Kinking, crushing, or any other damage resulting in distortion of the rope.

   d. Evidence of any type of heat damage.

   e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8" to 1 1/4" inclusive.

2. Prior to operations in the RESTRICTED MODE

   a. the controlled area limit switches will be tested;

   b. the "two-block" limit switches will be tested;

   c. the "inching hoist" controls will be tested.

3. The empty spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to any series of fuel cask handling operations.

3.10 BASES (Cont'd)

D. Time Limitation

The radiological consequences of a fuel handling accident are based upon the accident occurring at least 24 hours after reactor shutdown.

E. Standby Gas Treatment System

Only one of the two Standby Gas Treatment subsystems is needed to clean up the reactor building atmosphere upon containment isolation. If one subsystem is found to be inoperable, there is no immediate threat to the containment system performance and refueling operation may continue while repairs are being made. If both subsystems are inoperable, the plant is brought to a condition where the Standby Gas Treatment System is not required.

F. Core Standby Cooling Systems

During refueling the system cannot be pressurized, so only the potential need for core flooding exists and the specified combination of the Core Spray or LPCI subsystems can provide this. A more detailed discussion is contained in the bases for 3.5.F.

G. Control Room Emergency Filter System

If the system is found to be inoperable, there is no immediate threat to the control room and refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, refueling operations will be terminated.

H. Spent Fuel Cask Handling

The operation of the redundant crane in the Restricted Mode during fuel cask handling operations assures that the cask remains within the controlled area once it has been removed from its transport vehicle (i.e., once it is above the 931' elevation). Handling of the cask on the Refueling Floor in the Unrestricted Mode is allowed only in the case of equipment failures or emergency conditions when the cask is already suspended. The Unrestricted Mode of operation is allowed only to the extent necessary to get the cask to a suitable stationary position so the required repairs can be made. Operation with a failed controlled area microswitch will be allowed for a 48-hour period providing an Operator is on the floor in addition to the crane operator to assure that the cask handling is limited to the controlled area as marked on the floor. This will allow adequate time to make repairs but still will not restrict cask handling operations unduly.

4.10 BASES

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

LIMITING CONDITIONS FOR OPERATION

3.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the operating status of the Control Room Emergency Filter system, the Reactor Equipment Cooling system and the Service Water system.

Objective:

To assure the availability of the Control Room Emergency Filter system, the Reactor Equipment Cooling system and the Service Water system upon the conditions for which the capability is an essential response to station abnormalities.

SURVEILLANCE REQUIREMENTS

4.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the surveillance requirements for the Control Room Emergency Cooling system, the Reactor Equipment Cooling system and the Service Water system which are required by the corresponding Limiting Conditions for Operation.

Objective:

To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
A. Control Room Emergency Filter System	A. Control Room Emergency Filter System
1. Except as specified in Specification 3.12.A.3 below, the Control Room Emergency Filter system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.	1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal absorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.
2.a. The results of the in-place cold DOP leak tests on the HEPA filters shall show ≥ 99% DOP removal. The results of the halogenated hydrocarbon leak tests on the charcoal adsorbers shall show ≥99% halogenated hydrocarbon removal. The DOP and halogenated hydrocarbon tests shall be performed at a flowrate of ≤ 341 CFM.	2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once every 18 months for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show ≥99% radioactive methyl iodide removal with inlet conditions of: velocity ≥22 FPM, ≥1.75 mg/m^3 inlet iodide concentration, ≥ 95% R.H. and ≤30°C.	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. The emergency bypass fan shall be shown to provide 341 CFM ±10%.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.
	d. The system shall be operated at least 10 hours every month.
3. From and after the date that the Control Room Emergency Filter system is made or found to be inoperable for any reason, reactor operations are permissible only during the succeeding seven days unless the system is sooner made operable. Refueling requirements are as specified in Specification 3.10.G.	3. At least once per operating cycle automatic initiation of the system shall be demonstrated.
4. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours.	

3.12 BASES

A. Control Room Emergency Filter System

The Control Room Emergency Filter system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and HEPA filters. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the performance of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours.

B. Reactor Equipment Cooling (REC) System

The Reactor Equipment Cooling System consists of two, distinct subsystems, each containing two pumps and one heat exchanger. Each subsystem is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either subsystem.

The REC System has additional flexibility provided by the capability of interconnection of the two subsystems and the backup water supply to the critical cooling loop by the Service Water System. This flexibility and the need for only one pump in one critical cooling loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of the LPCI or Core Spray systems.

C. Service Water System

The Service Water System consists of two, distinct subsystems, each containing two vertical Service Water pumps located in the intake structure, and associated strainers, piping, valving and instrumentation. The pumps discharge to a common header from which independent piping supplies two Seismic Class I cooling water loops and one turbine building loop. Automatic valving is provided to shutoff all supply to the turbine building loop on drop in header pressure thus assuring supply to the Seismic Class I loops each of which feeds one diesel generator, two RHR Service Water booster pumps, one control room basement fan coil unit and one REC heat exchanger. Valves are included in the common discharge header to permit the Seismic Class I Service Water System to be operated as two independent subsystems. The heat exchangers are valved such that they can be individually backwashed without interrupting system operation.

3.12 BASES (cont'd)

During normal operation two or three pumps will be required. Three pumps are used for a normal shutdown.

The loss of all a-c power will trip all operating Service Water pumps. The automatic emergency diesel generator start system and emergency equipment starting sequence will then start one selected Service Water pump in 30-40 seconds. In the meantime, the drop in Service Water header pressure will close the turbine building cooling water isolation valve guaranteeing supply to the reactor building, the control room basement, and the diesel generators from the one Service Water pump.

Due to the redundance of pumps and the requirement of only one to meet the accident requirements, the 30 day repair time is justified.

D. Battery Room Ventilation

The temperature rise and hydrogen buildup in the battery rooms without adequate ventilation is such that continuous safe operation of equipment in these rooms cannot be assured.

4.12 BASES

A. Control Room Emergency Filter System

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant should be performed in accordance with ANSI N510-1980.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The test canisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 5.1 of ANSI N509-1980. The replacement tray for the absorber 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 HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.-ber.

4.12 BASES (cont'd)

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 is necessary to assure system performance capability.

B. Reactor Equipment Cooling System

Normal plant operation requires one heat exchanger and three pumps. Therefore, normal equipment rotation will demonstrate pump operability.

Pump rates will be demonstrated every three months as an indication of the pump condition.

C. Service Water System

The Service Water pumps shall be proven operable by their use during normal station operations. Since three pumps are continuously operating during normal operation and only one pump is required during accidents, the normal equipment rotation shall prove the pump operability.

Pump discharge head tests will be run every three months to verify the pumping ability.

Any silting problems caused by the Service Water system will be analyzed during and following the Preoperational Test Program. Any required changes in operating procedures, technical specifications or surveillance requirements will be made prior to CNS commercial operation.

D. Battery Room Ventilation

The ventilation fans will be rotated on a weekly basis to demonstrate operability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated September 9, 1992, the Nebraska Public Power District (the licensee) submitted a request for changes to the Cooper Nuclear Station (CNS) Technical Specifications (TS). The requested changes modify TS Table 3.1.1, Reactor Protection System Instrumentation Requirements, to indicate that the Main Steam Line Isolation Valve (MSIV) Closure trip is only required while the plant is in the RUN mode of operation. Also, the amendment adds bases for the instrument settings for the Core Spray and the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) systems to the TS Bases section for the Core Standby Cooling System (CSCS). The amendment also changes TS Limiting Condition for Operation (LCO) 3.12.A.2.c to clarify that only one emergency bypass fan exists in the Control Room Emergency Filter system. Finally, minor editorial changes and typographical error corrections are made to TS pages iii, 48, 63, 78, 87, 206, 209a, 215, 215a, 215d, 215e, and 215f.

The specific changes made to the TS are as follows:

(1) On page 29, Table 3.1.1, Reactor Protection System Instrumentation Requirements, is revised by deleting the "X" and its associated footnotes indicating that the MSIV closure trip is applicable while the plant is in the STARTUP mode. The modified table indicates that this trip is applicable only when the mode selector switch is in the RUN position.

(2) On page 85, the following Bases for the Core Spray and RHR (LPCI mode) instrument settings are added:

"CORE SPRAY

Initiation and control instrumentation settings ensure that the Core Spray system operates to ensure that peak fuel element cladding temperatures do not exceed 2200°F during a design basis LOCA. The basis for the settings is discussed in USAR [Updated Safety Analysis Report] Section VII-4.

RESIDUAL HEAT REMOVAL (LPCI MODE)

Initiation and control instrumentation settings ensure that the LPCI mode of the Residual Heat Removal system operates to ensure that peak fuel element cladding temperatures do not exceed 2200°F during a design basis LOCA [Loss-of-Coolant Accident]. High drywell pressure and reactor water level instrumentation also allow injection water to be diverted for containment spray. The basis for the settings is discussed in USAR Section VII-4."

(3) On pages 215 and 215f, the term "Reactor Building Closed Cooling Water System" is replaced by the term "Reactor Equipment Cooling System." This change clarifies the existing TS requirements by utilizing a new term that is consistent with the terminology used in Amendment No. 152.

(4) On pages iii, 48, 63, 78, 87, 206, 209a, 215, 215a, 215d, and 215e, the terms "Main Control Room Ventilation Isolation System," "Main Control Room Ventilation System," and "Control Room Air Treatment System" are replaced by the term "Control Room Emergency Filter System." This change clarifies the existing TS requirements through the use of a single term for this system that is consistent with the system descriptions given in CNS USAR Section X-10.3.6.5 and in the Standard Technical Specifications.

(5) On page 215a, LCO 3.12.A.2.c has been revised to clearly identify that this LCO is applicable to the emergency bypass fan, as opposed to each fan. This change clarifies in the previous wording, which implied that more than one fan exists. The change correctly reflects the current configuration of the plant. There are no changes to the configuration of the Control Room Emergency Filter system associated with this change.

(6) On page iii, a page reference pertaining to Section 3.12.A is changed and the subsection headings associated with Section 5.0 are moved to their proper location.

(7) On page 48, parentheses are removed from LCO 3.2.D.5.b.

(8) On page 63, the term "Isolation (4)" is moved under the first column and parentheses are removed from the term "RMV-RM-1."

(9) On page 78, the comma is removed from the term "RMP-RM-251 A, B, C & D."

(10) On page 206, a Surveillance Requirement is renumbered from "3.10" to "4.10."

(11) On page 215a, Surveillance Requirement 4.12.A.2.d is appropriately numbered, the referenced specification is changed to "3.12.A.3" in LCO 3.12.A.1, and in LCO 3.12.A.3, the spelling of "succeeding" is corrected.

(12) On page 215f, the term "Service Water" is capitalized.

(13) To improve clarity, all of the LCOs under 3.12.A and the associated surveillance requirements under 4.12A are moved to page 215a. For similar reasons, a portion of BASES 3.12.C is moved from page 215e to page 215d.

2.0 EVALUATION

With regard to the change in TS Table 3.1.1, the licensee stated that Amendment No. 83 to the CNS operating license authorized the changes to the TS needed for the implementation of a plant modification which added the Low Low Set function to the safety relief valves. This plant modification is described in the licensee's letters to the NRC dated December 17, 1982, and February 15, 1983. The licensee stated that, during a recent operator licensing examination, it was noted that these changes failed to include the modification performed to the circuitry which bypasses the MSIV Closure trip when the reactor mode selector switch is not in the RUN mode. Prior to Amendment No. 83, this trip was applicable when the mode selector switch was also in the STARTUP position, with bypass controlled by four pressure switches. The function of these pressure switches was reassigned to the Low Low Set function by the Low Low Set plant modification, and the MSIV Closure trip circuitry in the STARTUP mode eliminated. However, Table 3.1.1 was not updated to reflect the changes to the MSIV Closure trip circuitry which were approved by Amendment No. 83. The changes made by this amendment correct this oversight made by the licensee in proposing the TS changes associated with Amendment No. 83, and make the Table 3.1.1 consistent with the plant modifications previously approved by Amendment No. No. 83. On this basis the NRC staff finds the proposed changes to TS Table 3.1.1 to be acceptable.

The licensee also stated that the changes to the TS Bases that were made in Amendment No. 83 included Bases section for the new Low Low Set instrumentation settings. However, the changes proposed by the licensee and implemented by Amendment No. 83 erroneously indicated that the settings for the Core Spray and RHR (LPCI mode) system instruments had "No Basis." The change made by this amendment adds appropriate bases for the Core Spray and RHR system instrumentation. The licensee proposed that the Bases for the instrument tables for these two systems reference Updated Safety Analysis Report (USAR) Section VII-4, which describes the Core Spray and RHR (LPCI mode) system initiating and control instrument settings, and that operation of the systems ensures that fuel cladding temperatures will not exceed 2200°F during design basis loss-of-coolant accidents. Based on its review of the proposed new Bases, the staff finds them acceptable because they correct an error made at the time Amendment No. 83 was requested.

The other changes have been reviewed by the NRC staff and found acceptable because they are administrative changes that improve the consistency of the TS and correct a variety of minor errors.

In summary, the NRC staff has reviewed the changes made by this amendment and finds that all the changes are administrative in nature and are made to clarify the TS, to make the TS consistent with the plant configuration, to make the TS internally consistent, and to correct minor errors in the TS. On this basis the staff finds them 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

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 61114). 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.

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.

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Date: April 23, 1993