September 25, 1986

Docket No. 50-298

Mr. J. M. Pilant, Technical
  Staff Manager
Nuclear Power Group
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68601

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment is in response to your application dated April 26, 1985, (Change No. 18) as supplemented by letters dated May 24, 1985, June 14, 1985, and July 3, 1986.

The amendment changes the Technical Specifications in the following areas: (1) Standby Gas Treatment and Control Room Ventilation Systems, (2) Sample line isolation setpoint change (3) Refueling interlocks (4) Typographical errors (5) Environmental Qualification deadline, and (6) Table of Contents correction.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's Bi-Weekly Federal Register notice.

Sincerely,

Original signed by

William O. Long, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 102 to
   License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. J. M. Pilant
Nebraska Public Power District

Cooper Nuclear Station

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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. 102
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated April 26, 1985, as supplemented by submittals dated May 24, 1985, and June 14, 1985, and July 3, 1986 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 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 licensee 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 Specification

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 25, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

1. Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines

Pages
iv
50
52b
75
83
111
165
182
183
205
215
215d
221
222
223
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COOPER NUCLEAR STATION
TABLE 3.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured (2)
Main Steam Line High Rad.	RMP-RM-251, A,B,C,&D	≤ 3 Times Full Power	2	A or B
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D #1	≥+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Low Water Level	NBI-LIS-57 A & B #1 NBI-LIS-58 A & B #1	≥-145.5" Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	≤ 200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,&D 117, 118, 119	≤ 150% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134, A,B,C,&D	≥ 825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,&D	≤ 2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	≤ 75 psig	1	D
Main Condenser Low Vacuum	MS-PS-103, A,B,C,&D	≥ 7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	≤ 200% of System Flow	1	C

Amendments Nos. 67, 88, 96, 100, 101, 102

NOTES FOR TABLE 3.2.A (cont'd.)

Isolations

1. Secondary Containment Isolation
2. Start Standby Gas Treatment System

Group 7

Isolation Signals:

1. Reactor Low Low Low Water Level ($\geq$-145.5 in)
2. Main Steam Line High Radiation ($\leq$3 times full power background)

Isolations:

1. Reactor Water Sample Valves

Amendments Nos. ~~55~~, ~~58~~, ~~88~~, ~~101~~, 102

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 6)
RCIC TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
1. Reactor High Water Level	NBI-LIS-101 A & C, #2	Once/Month (1)	Once/3 Months	Once/Day
2. Reactor Low Water Level	10A - K79 A & B 10A-K80 A & B	Once/Month (1)	Once/3 Months	Once/Day
3. RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	Once/Month (1)	Once/3 Months	None
4. RCIC Low Pump Suction Press.	RCIC-PS-67-1	Once/Month (1)	Once/3 Months	None
5. RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-80, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-81, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-82, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
6. RCIC Steam Line High ΔP	RCIC-dPIS-83	Once/Month (1)	Once/3 Months	None
	RCIC-dPIS-84	Once/Month (1)	Once/3 Months	None
7. RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C, & D	Once/Month (1)	Once/3 Months	None
8. RCIC Low Pump Disch. Flow	RCIC-FIS-57	Once/Month (1)	Once/3 Months	None
9. Pump Disch. Line Low Pressure	CM-PS-269	Once/3 Months	Once/3 Months	None
10. RCIC Turbine Conditional Supv. Alarm Timer	RCIC-TDR - K9	Once/Month (1)	Once/Oper. Cycle	None
11. RCIC Steam Line High ΔP Actuation Timer	RCIC-TDR-K-12	Once/Month	Once/Oper. Cycle	None
	RCIC-TDR-K-32	Once/Month	Once/Oper. Cycle	None
Logic Systems (4)(6)				
1. Logic Bus Power Monitor		Once/6 Months	N.A.	
2. RCIC Initiation		Once/6 Months	N.A.	
3. Turbine Trip		Once/6 Months	N.A.	
4. RCIC Automatic Isolation		Once/6 Months	N.A.	

Amendments Nos. ~~75~~, ~~80~~, ~~101~~, 102

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance, and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

A. Primary Containment Isolation Functions

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation, set to trip at 176.5" (+12.5") above the top of the active fuel, closes all isolation valves except those in Groups 1, 4, 5, and 7. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low low reactor water level instrumentation is set to trip when the water level is 19" (-145.5") above the top of the active fuel. This trip closes Groups 1 and 7 Isolation Valves (Reference 1), activates the remainder of the CSCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished, The volume versus concentration requirement of the solution is such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature versus concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shutdown the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 2650 gallons of solution having a 16 percent sodium pentaborate concentration, or the equivalent as shown in Figure 3.4.1, is required to meet this shutdown requirement. For the minimum required pumping rate of 38.2 gpm, the maximum net storage volume of the boron solution is established as 4780 gallons.

4.4 BASES

STANDBY LIQUID CONTROL SYSTEM

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The bases for the surveillance requirements are given in subsection III.9.6 of the Final Safety Analysis Report, and the details of the various tests are discussed in subsection III.9.5. The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7. (cont'd.)

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both standby gas treatment systems shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon leak tests at $\leq$ design flow (1780 CFM) and at a reactor building pressure $\leq$-.25" Wg on HEPA filters and charcoal adsorber banks respectively shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$99% radioactive methyl iodide removal with inlet conditions of: velocity $\geq$42 FPM, $\geq$1.75 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\leq 30^{\circ}C$.

c. Each fan shall be shown to provide 1780 CFM ±10%.

3. From and after the date that one standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the other standby gas treatment system, and its associated diesel generator, shall be operable.

Fuel handling requirements are specified in Specification 3.10.E.

4.7 (cont'd.)

B. Standby Gas Treatment System

1. At least once per operating cycle the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.

b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year 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. 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. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each system shall be operated with the heaters on at least 10 hours every month.

e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. System drains where present shall be inspected quarterly for adequate water level in loop-seals.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. 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 10 CFR 100 guidelines for the accidents analyzed.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither system is operable, the plant is brought to a condition where the standby gas treatment system is not required.

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

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. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with ANSI N510-1980. The test cannisters 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 1 of Regulatory Guide 1.52, Revision 2, March, 1978. 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 system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system 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.

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.10.A (Cont'd)

6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. When fuel is present in the reactor vessel, all other refueling interlocks shall be operable.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

2. Operable SRM's shall have a minimum of 3 cps except as specified in 3 and 4 below.

3. Prior to spiral unloading, the SRM's shall have an initial count rate of 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.

4.10 (Cont'd)

B. Core Monitoring

Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response (or every 12 hours until 3 cps is attained if the spiral reload technique is being used).

3.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the operating status of the main control room ventilation system, the reactor building closed cooling water system and the service water system.

Objective:

To assure the availability of the main control room ventilation system, the reactor building closed cooling water system and the service water system upon the conditions for which the capability is an essential response to station abnormalities.

A. Main Control Room Ventilation

1. Except as specified in Specification 3.12.A.3 below, the control room air treatment 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.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\leq$ design flow (341 CFM) and at control room pressure on HEPA filters and charcoal adsorber banks respectively shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$ 99% radioactive methyl iodide removal with inlet conditions of: velocity $\geq$22 FPM, $\geq$1.75 mg/m^3 inlet iodide concentration, $\geq$ 95% R.H. and $\leq$30°C.

c. Each fan shall be shown to provide 341 CFM ±10%.

4.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water 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.

A. Main Control Room Ventilation

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 tests and sample analysis of Specification 3.12.A.2 shall be performed at least once per year 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. 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. 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.

A. Main Control Room Ventilation System

The control room ventilation 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 Building Closed Cooling Water System

The reactor building closed cooling water system has two pumps and one heat exchanger in each of two loops. Each loop is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either loop.

The system has additional flexibility provided by the capability of inter-connection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one 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 LPCI or the core spray systems.

C. Service Water System

The service water system consists of four 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 RBCCW

6.2 (cont'd)

f. Investigate all violations of Technical Specifications, including reporting evaluation and recommendations to prevent recurrence, to the Vice President - Nuclear and to the Chairman of the NPPD Safety Review and Audit Board.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review and Audit Board.

h. Review all reportable events specified in Section 50.73 to 10CFR Part 50.

i. Review drills on emergency procedures (including plant evacuation) and adequacy of communication with off site groups.

j. Periodically review procedures required by Specifications 6.3.1, 6.3.2, 6.3.3, and 6.3.4 as set forth in administrative procedures.

5. Authority

a. The Station Operations Review Committee shall be advisory.

b. The Station Operations Review Committee shall recommend to the Division Manager of Nuclear Operations approval or disapproval of proposals under items 4, a through e and j above. In case of disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations, the course determined by the Division Manager of Nuclear Operations to be the more conservative will be followed. A written summary of the disagreement will be sent to the Vice President - Nuclear and to the NPPD Safety Review and Audit Board.

c. The Station Operations Review Committee shall report to the Chairman of the NPPD Safety Review and Audit Board on all reviews and investigations conducted under items 4.f, 4.g, 4.h, and 4.i.

d. The Station Operations Review Committee shall make determinations regarding whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the NPPD Safety Review and Audit Board.

6. Records:

Minutes shall be kept for all meetings of the Station Operations Review Committee and shall include identification of all documentary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Vice President - Nuclear within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. NPPD Safety Review and Audit Board (SRAB)

Function: The Board shall function to provide independent review and audit of designated activities.

1. Membership:

   a. Chairman

   b. Vice-Chairman

   c. Five Members

   d. Consultants (as required)

   The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, quality assurance practices, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized.

   Alternate members shall be appointed in writing by the Board Chairman to serve on a temporary basis; however, no more than two alternates shall serve on the Board at any one time.

2. Meeting frequency: Semiannually, and as required on call of the Chairman.

3. Quorum: Chairman or Vice Chairman, plus four members including alternates. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Review: The following subjects shall be reported to and reviewed by the NPPD Safety Review and Audit Board.

   a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

   b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or the CNS Operating License.

e. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All reportable events specified in Section 50.73 to 10CFR Part 50.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Minutes of meetings of the Station Operations Review Committee.

j. Disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations.

k. Review of events covered under e,f,g, and h above include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority: The NPPD Safety Review and Audit Board shall report to and be advisory to the Vice President - Nuclear on those areas of responsibility specified in Specifications 6.2.1.B.4 and 6.2.1.B.7.

6. Records:

Minutes shall be recorded for all meetings of the NPPD Safety Review and Audit Board and shall identify all documentary material reviewed. Copies of the minutes shall be forwarded to the Vice President - Nuclear and the Division Manager of Nuclear Operations, and such others as the Chairman may designate within one month of the meeting.

7. Audits:

Audits of selected aspects of plant operation shall be performed under the cognizance of SRAB with a frequency commensurate with their safety significance. Audits performed by the Quality Assurance Department which meet this specification shall be considered to meet the SRAB audit requirements if the audit results are reviewed by SRAB. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. These audits shall encompass:

a. Verification of compliance with internal rules, procedures (for example: normal, off-normal, emergency, operating, maintenance, surveillance, test, and radiation control procedures) and applicable license conditions at least once per 24 months.

b. The training, qualification, and performance of the operating staff at least once per 24 months.

c. The Emergency Plan and implementing procedures at least once per 12 months.

d. The Security Plan and implementing procedures at least once per 12 months.

e. The facility fire protection and its implementing procedures at least once per 24 months.

f. A fire protection and loss prevention inspection will be performed utilizing either qualified off-site licensee personnel or an outside fire protection consultant at least once per 12 months.

g. An inspection and audit by an outside qualified fire protection consultant shall be performed at least once per 36 months.

h. The Radiological Environmental Monitoring Program and the Offsite Dose Assessment Manual with their implementing procedures at least once every 24 months.

6.3 PROCEDURES AND PROGRAMS

6.3.1 Introduction

Station personnel shall be provided detailed written procedures to be used for operation and maintenance of system components and systems that could have an effect on nuclear safety.

6.3.2 Procedures

Written procedures and instructions including applicable check off lists shall be established, implemented, and maintained for the following:

A. Normal startup, operation, shutdown and fuel handling operations of the station including all systems and components involving nuclear safety.

B. Actions to be taken to correct specific and foreseen potential or actual malfunctions of safety related systems or components including responses to alarms, primary system leaks and abnormal reactivity changes.

C. Emergency conditions involving possible or actual releases of radioactive materials.

D. Implementing procedures of the Security Plan and the Emergency Plan.

E. Implementing procedures for the fire protection program.

F. Administrative procedures for shift overtime.

G. Implementing procedures for the Offsite Dose Assessment Manual.

6.3.3 Maintenance and Test Procedures

The following maintenance and test procedures will be provided to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

A. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

B. Routine testing of standby and redundant equipment.

C. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

D. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

E. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

6.3.4 Radiation Control Procedures

Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR 20.

A. High Radiation Areas

In lieu of the "control device" or "alarm signal" required by Paragraph 20.203 (c) (2) of 10 CFR 20 each High Radiation Area (100 mrem/hr or greater) shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring notification and permission of the shift supervisor. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

6.3.5 Temporary Changes

Temporary changes to procedures which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the operating staff holding SRO licenses. Such changes shall be documented and subsequently reviewed by the Division Manager of Nuclear Operations within one month.

6.3.6 Exercise of Procedures

Drills of the Emergency Plan procedures shall be conducted annually, including a check of communications with offsite support groups. Drills on the procedures specified in 6.3.2.A, B, and C above shall be conducted as part of the retraining program.

6.3.7 Programs

The following programs shall be established:

A. Systems Integrity Monitoring Program

A program shall be established to reduce leakage to as low as practical levels from systems outside the primary containment during a serious accident that would or could contain highly radioactive fluids. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

B. Iodine Monitoring Program

A program shall be established to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

C. Environmental Qualification Program

A. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

D. Post-Accident Sampling System (PASS)

A program shall be established to ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. This program shall include training of personnel, procedures for sampling and analysis and provisions for operability of sampling and analysis equipment.

6.5.2 Reportable Events

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50. The NRC shall be notified and a report submitted pursuant to the requirements of Section 50.73. Each Reportable Event shall be reviewed by SORC and the results of this review shall be submitted to SRAB and the Vice President - Nuclear.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO.50-298

1.0 INTRODUCTION

By letters dated April 26, 1985, May 24, 1985, June 14, 1985 and July 3, 1986 the Nebraska Public Power District (the licensee) requested an amendment to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The proposed amendment would change the Technical Specifications in the following areas: (1) Standby Gas Treatment System (SGTS) and Control Room Ventilation System (CRVS) operability and surveillance requirements: (2) Reactor Water Sample Line Isolation trip setting; (3) Refueling Interlocks requirements; (4) Equipment Qualification (EQ) deadline; (5) Typographical errors, and (6) Table of Contents corrections.

2.0 DISCUSSION AND EVALUATION

SGTS and CRVS

The licensee has requested changes to the Technical Specifications applicable to these systems as follows: (1) Section 3.7.B.2.a would be changed to clarify that the in-place leak tests on HEPA filters and charcoal adsorbers of the Standby Gas Treatment System shall be conducted at equal to or less than design flow (1780 CFM) and at a reactor building pressure equal to or less than -0.25 inch water gauge. The existing specification requires that the tests be conducted "at design flow." (2) Section 3.7.B.2.b would be changed to clarify that the SGTS carbon sample laboratory analysis be conducted with an inlet velocity of equal to or greater than 42 FPM. The existing specification requires that the analysis be conducted "at a velocity within 20 percent of actual systems design." (3) Section 3.7.B.2.c would be changed to clarify that each SGTS fan be shown to provide 1780 CFM plus or minus 10%. The existing specification requires that each fan be shown to "operate within + 10% of design flow." (4) The Bases for Sections 3.7.B and 3.7.C would be changed to clarify that the SGTS in-place tests should indicate a HEPA filter efficiency of at least 99 percent removal of DOP. It would be deleted that operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and the charcoal adsorbers. (5) Bases for Sections 4.7.B and 4.7.C would be changed to clarify the references to Regulatory Guide 1.52, Revision 2, March 1978. (6) Section 3.12.A.2.a would be changed to clarify that the in-place leak tests on the HEPA filters and the charcoal adsorbers of the Main Control Room Ventilation System shall be conducted at equal to less than design flow (341 CFM) and at control room pressure. The existing specification requires that the tests be conducted "at design flows." (7) Section 3.12.A.2.b would be changed to clarify that the CRVS laboratory carbon sample analysis be conducted with an inlet velocity of equal to or greater than 22 feet per minute. The existing specification requires that the anlysis be conducted "at a velocity within 20 percent of system design." (8) Section 3.12.B.2.c would be changed to clarify that each CRVS fan be shown to provide 341 CFM plus or minus 10%. The existing specifications requires each fan to "operate within plus or minus 10% of design flow." (9) Bases for Section 3.12.A would be changed to clarify that the in-place tests should indicate a HEPA filter efficiency of at least 99 percent removal of DOP. It would be deleted that operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and the charcoal adsorbers.

The Standard Technical Specifications for General Electric BWRs (NUREG-0123) specifies that for in-place testing of ESF filter systems, such as for the SGTS and the CRVS, the system flow rate be the design flow (appropriate value given) plus or minus 10%. There are no provisions governing flow conditions under which the carbon samples are analyzed other than by reference to Regulatory Guide 1.52, which states that the sample should be exposed to the same service conditions as the adsorber section. There is a provision for verifying the system flow rate to be (appropriate design value given) plus or minus 10%. Regulatory Guide 1.52, Revision 2, in each of the provisions under "In-Place Testing Criteria" refers to ANSI-N510-1975. The significance of the in-place leak tests is addressed in Appendix B of ANSI-N510-1975. The in-place field tests of installed HEPAs do not show the efficiencies of the filters but only reveal the presence of leaks in the system. The in-place field tests of installed adsorbers are designed to determine only the amount of leakage through or around the installed bank of cells. Poor HEPA filter or charcoal adsorber performance is not detected by these tests.

With HEPAs, it can be inferred that the particle-removing efficiency of the system is equivalent to that of the individual filters if penetration observed in the in-place test is equivalent to the penetration established during factory testing of the individual filters. For adsorbers, true efficiency tests are run on small representative samples of adsorbent. An installed system can be assumed to have an efficiency equivalent to that of the sample only if (1) the laboratory sample is representative, (2) the adsorber cells are tightly packed, and (3) there are no leaks or bypasses in the factory test of leakage through or around the adsorbent in the cell or the in-place test of leakage through or around the installed bank of cells.

While particle collection efficiencies and charcoal adsorbent efficiencies may be highly dependent on the airflow rate, the leakage rate relative to the airflow rate should not vary significantly with the airflow rate for a dimensionally stable system. ANSI-N510-1975 does not prescribe an airflow rate for the in-place leak tests for either the HEPA filter banks or the installed adsorber stage. ERDA 76-21, "Nuclear Air Cleaning Handbook," referred to by Regulatory Guide 1.52, states that for HEPA filters the in-place tests can be made at rated system airflow or at reduced flow. Some agencies test at as low as 5 to 10% of rated system airflow. The The licensee has reviewed the Cooper SGTS and CRVS filter and filter housing designs. In a letter dated July 3, 1986, the licensee verified that they are dimensionally stable and have no loop seals or other features that could allow higher bypass leakage rates at design airflow than would be determined by in-place leak tests at any lower airflow rate. Therefore, the changes (1) and (6) to Sections 3.7.B.2.a and 3.12.A.2.a to conduct the in-place leak test at equal to or less than design flow are acceptable.

The licensee's submittal states that the filterface velocity of 42 FPM for the SGTS corresponds to the design flow rate of 1,780 CFM and that the filterface velocity of 22 FPM for the CRVS corresponds to the design flow rate of 341 CFM. Laboratory testing of carbon samples at inlet velocities equal to or in excess of the above filterface velocities is acceptable since residence times during testing will not be in excess of in-service residence times and, therefore, the carbon filter efficiencies obtained during laboratory testing will be expected to be equaled or exceeded in service. Therefore, the changes (2) and (7) to Sections 3.7.B.2.b and 3.12.A.2.b to conduct laboratory carbon sample analyses at inlet velocities corresponding to design flow rates are acceptable.

The other proposed changes: (3), (4), (5), (8) and (9) are of an editorial nature, are consistent with current requirements and are also acceptable.

Reactor Water Sample Line Water Level Trip Setting

The reactor water sample line primary containment isolation valves are NORMALLY CLOSED air operated valves. They are required by Technical Specifications to automatically isolate on a high main steam line radiation signal or a low-low (-37 in.) reactor vessel water level signal. The licensee has requested that the Technical Specifications be changed to specify that the latter (reactor water level isolation function) occur at the low-low-low (-145.5 in.) setpoint. The effect of the setpoint reduction would be an additional delay in sample line isolation should a reduction in reactor vessel inventory event occur. However, in the case of a small break outside containment, such as the 3/4-inch sample line, no significant reduction in vessel level would occur. The event would be terminated by manual isolation on other indications (i.e. high secondary containment radiation levels or temperature levels, increased reactor makeup water requirements, or visual observation by roving patrols) as indicated in the applicable protective action sequence (ref. Cooper USAR Appendix G). Therefore, changing the reactor vessel level setpoint for sample line isolation would not adversely affect the consequences of a sample line break. The amendment is therefore acceptable.

Refueling Interlocks

The current CNS Technical Specifications require that all refueling interlocks, with the exception of the one rod out interlock, be operable during multiple control rod removal regardless whether fuel is or is not in the vessel. The amendment would delete this requirement for periods when there is no fuel in the reactor vessel. The objective of the interlocks, as stated in the Technical Specifications, is to ensure that reactivity control is within the capability of the control rods and to prevent inadvertent criticality during refueling operations. With no fuel in the vessel no core reactivity is available. The change is therefore acceptable.

Environmental Qualification Deadline

Section 6.3.7.C of the CNS Technical Specifications presently contains a June 30, 1982 EQ deadline. That requirement was placed in the CNS Technical Specifications by an Order dated October 24, 1980. However, the June 30, 1982 deadline was superseded by 10 CFR 50.49(g). The licensee requests that 6.3.7.C be deleted. With the Rule in effect, this would be a purely administrative change and would have no effect on the actual EQ program requirements. The change is therefore acceptable.

Typographical Errors

To correct typographical errors, the licensee requests the following changes (1) In 4.4 Bases, Standby Liquid Control System change "III.8.5" to "III.9.5"; and (2) In Table 4.2.B (Page 6) change "Logic Buss Power Monitor" to "Logic Bus Power Monitor". These changes have no safety significance and are acceptable.

Reformat of Administrative Controls Subsections

Section 6, "Administrative Controls" of the CNS Technical Specifications has undergone various revisions which have introduced discontinuities between the pages of some subsections. The licensee proposes to make editorial changes which condense related subsections. The content of the material would not be changed. Page 225a would be deleted as a result of compressing its contents into fewer pages. The proposed change would improve readability and comprehension of the Technical Specifications and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This 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. 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. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

4.0 CONCLUSION

We have concluded, based in 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and 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 Contributer: W. Long, L. Lois, C. Nichols, J. Lane

Dated: September 25, 1986