March 2, 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. 158 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. M83786)

The Commission has issued the enclosed Amendment No. 158to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The amendment consists of changes to the CNS Technical Specifications (TS) in response to your application dated May 4, 1992, as supplemented by letters dated October 15, 1992, and January 13, February 12, and February 24, 1993.

The amendment authorizes removal of the Main Steam Line Radiation Monitor (MSLRM) scram and Group I Containment Isolation functions, and modifies the TS accordingly.

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

Sincerely,
/S/
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. 158 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

* See previous concurrence

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OFC	LA:PD4-1	PM:PD4-1	BC:SRXB	OGC	D(A):PD4-1
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 2, 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

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Sincerely,

Harry Rood

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

cc w/enclosures:
See next page

cc:

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Columbus, Nebraska 68602-0499

Cooper Nuclear Station
ATTN: Mr. John M. Meacham
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Mr. Richard Moody, Chairman
Nemaha County Board of Commissioners
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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. 158
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated May 4, 1992, as supplemented by letters dated October 15, 1992, and January 13, February 12, and February 24, 1993, 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 158, 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

George Hubbard

George T. Hubbard, 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: March 2, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 158

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
29	29
30	30
33	33
34	34
35	35
36	36
39	39
48	48
50	50
52	52
63a	63a
68	68
78	78
81	81
84	84

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)(9)		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

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power to less than 30% of rated.

   C. Reduce power level to IRM range and place mode switch in the Startup position within 8 hours and depressurize to less than 1000 psig.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. This note deleted.

4. Permissible to bypass when turbine first stage pressure is less than 30% of full load.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a full scram being initiated.

7. When the reactor is subcritical, fuel is in the vessel, and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode switch in shutdown.
   b. Manual scram.
   c. IRM high flux. 120/125 indicated scale.
   d. APRM (15%) high flux scram.

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

COOPER NUCLEAR STATION
TABLE 4.1.1 (Page 2)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level in Scram Discharge Volume CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D	A	Trip Channel and Alarm	Once/3 Months
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Turbine Control Valve Fast Closure TGF-63/OPC -1,2,3,4	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive MS-PS-14 A,B,C, & D	A	Trip Channel and Alarm	Once/3 Months
Turbine Stop Valve Closure SVOS-1 (1), SVOS-1 (2) SVOS-2 (1), SVOS-2 (2)	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.1

1. Initially once per month until exposure (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months after review and approval of the NRC. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of CNS.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required when the systems are not required to be operable or are tripped. If reactor startups occur more frequently than once per week, the maximum functional test frequency need not exceed once per week.

   If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. Deleted.

5. Test RPS channel after maintenance.

6. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the monthly functional test program.

COOPER NUCLEAR STATION
TABLE 4.1.2
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration Test (5)	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns (7)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/Week
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source (8)	Once/Refueling Outage
LPRM Signal	B	TIP System Traverse	Note (9)
High Reactor Pressure	A	Standard Pressure Source	Once/3 Months
High Drywell Pressure	A	Standard Pressure Source	Once/3 Months
Reactor Low Water Level	A	Pressure Standard	Once/3 Months
High Water Level in Scram Discharge Volume	A	Note (6)	Note (6)
Main Steam Line Isolation Valve Closure	A	Note (6)	Note (6)
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Once/6 Months
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Once/3 Months
Turbine Stop Valve Closure	A	Note (6)	Note (6)

NOTES FOR TABLES 4.1.2

1. A description of three groups is included in the bases of this Specification.

2. Calibration tests are not required when the systems are not required to be operable or are tripped but are required prior to return to service.

3. Deleted.

4. Maximum frequency required is once per week.

5. Response time is not a part of the routine instrument channel test, but will be checked once per operating cycle. The response time measurement will be the time segment from the time the sensor contacts actuate to the time the scram solenoid valves deenergize.

6. Physical inspection and actuation of these position switches will be performed during the refueling outages.

7. On controlled shutdowns, the IRM reading 120/125 of full scale will be set equal to or less than 45% of rated Power. All range scales above that scale on which the most recent IRM calibration was performed will be mechanically blocked.

8. The Flow Bias Scram Calibration will consist of calibrating the sensors, flow converters and signal offset networks during operation. The instrumentation is an analog type with redundant flow signals that can be compared. The flow bias trip and upscale will be functionally tested according to table 4.1.1 to assure proper operation during the operating cycle. Refer to Bases of 4.1 for further explanation of calibration frequencies.

9. LPRM detectors shall be calibrated every six weeks of reactor power operation above 20% of rated power.

3.1 BASES (Cont'd.)

initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core standby cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. paragraph VII.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The APRM (High flux in Start-Up or Refuel) system provides protection against excessive power levels and short reactor periods in the start-up and intermediate power ranges.

The IRM system provides protection

4.1 BASES (cont'd.)

2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T (M = nT).

3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1.1.

4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.

5. A test interval of 1 month will be used initially until a trend is established, which is based on system availability analysis and good engineering judgement plus operating experience.

Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purpose of analysis, it is assumed that this rare failure will be detected within two hours.

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are

(6) Reliability of Engineered Safety Features as a Function of Testing Frequency, I.M. Jacobs, "Nuclear Safety", Vol. 9, No. 4, July-Aug. 1968, pp. 310-312.

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. Main Control Room Ventilation Isolation

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.

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. Main Control Room Ventilation Isolation

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. |

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 Radiation	RMP-RM-251, A,B,C,&D	≤ 3 Times Full Power	2	E
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D #1	≥+4.5 in. 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 in. 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

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required there shall be two operable or tripped trip systems for each function.

2. If the minimum number of operable instrument channels per trip system requirement cannot be met by a trip system, that trip system shall be tripped. If the requirements cannot be met by both trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in a cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have the Main Steam Isolation Valves shut within 8 hours.

   C. Isolate the Reactor Water Cleanup System.

   D. Isolate the Shutdown Cooling mode of the RHR System.

   E. Isolate the Reactor Water Sample Valves.

3. Two required for each steam line.

4. These signals also start the Standby Gas Treatment System and initiate Secondary Containment isolation.

5. Not required in the refuel, shutdown, and startup/hot standby modes (interlocked with the mode switch).

6. Requires one channel from each physical location for each trip system.

7. Low vacuum isolation is bypassed when the turbine stop is not full open, manual bypass switches are in bypass and mode switch is not in RUN.

8. The instruments on this table produce primary containment and system isolations. The following listing groups the system signals and the system isolated.

   <u>Group 1</u>

   Isolation Signals:

   1. Reactor Low Low Low Water Level (≥-145.5 in.)
   2. Main Steam Line Low Pressure (≥825 psig in the RUN mode)
   3. Main Steam Line Leak Detection (≤200°F)
   4. Condenser Low Vacuum (≥7" Hg vacuum)
   5. Main Steam Line High Flow (≤150% of rated flow)

   Isolations:

   1. MSIV's
   2. Main Steam Line Drains

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

   A. (1) If radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for a period greater than 15 consecutive minutes, the off-gas isolation valve shall close and reactor shutdown shall be initiated immediately and the reactor placed in a cold shutdown condition within 24 hours.

   A. (2) Refer to Specification 3.21.A.2.

   B. A minimum of one instrument channel per trip system shall be operable when handling irradiated fuel inside secondary containment, and when moving loads inside secondary containment which have the potential to damage irradiated fuel. If this requirement cannot be met by a trip system, then that trip system shall be tripped. If this requirement cannot be met by both trip systems, then the following actions shall be taken:

      (1) Cease handling of irradiated fuel inside secondary containment and remove the load from over the irradiated fuel via the most direct path, or

      (2) Isolate secondary containment and start SBGT.

   C. During release of radioactive wastes, the effluent control monitor shall be set to alarm and automatically close the waste discharge valve prior to exceeding the limits of Specification 3.21.B.1.

   D. Refer to Section entitled "Additional Safety Related Plant Capabilities".

   E. Refer to Section 3.2.D.5 and the requirements for Primary Containment Isolation on high main steam line radiation, Table 3.2.A.

2. Trip settings to correspond to Specification 3.21.B.1.

3. Trip settings to correspond to Specification 3.21.C.6.a.

4. Minimum number of channels operable shall be one during mechanical vacuum pump operation.

Amendment No. ~~62,88,147~~, 158

COOPER NUCLEAR STATION
TABLE 4.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION SYSTEM
TEST AND CALIBRATION FREQUENCIES

Item	Item I.D. No.	Function Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Water Level	NBI-LIS-57, A & B #2 NBI-LIS-58, A & B #2	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Low Water Level	NBI-LIS-57, A & B #1 NBI-LIS-58, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
Main Steam Line High Radiation	RMP-RM-251, A,B,C,&D	Once/Month (1) (13)	Once/3 Months (14)	Once/Day
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	Once/Month (1)	Once/Operating Cycle	None
Main Steam Line High Flow	MS-dPIS-116, A,B,C,&D	Once/Month (1)	Once/3 Months	None
	117	Once/Month (1)	Once/3 Months	None
	118	Once/Month (1)	Once/3 Months	None
	119	Once/Month (1)	Once/3 Months	None
Main Steam Line Low Press.	MS-PS-134, A,B,C,&D	Once/Month (1)	Once/3 Months	None
High Reactor Pressure	RR-PS-128, A & B	Once/Month (1)	Once/3 Months	None
Condenser Low Vacuum	MS-PS-103, A,B,C,&D	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Flow	RWCU-dPIS-170, A & B	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Space Temp.	RWCU-TS-150 A-D, 151, 152, 153, 154, 155, 156, 157, 158, 159, RWCU-TS-81, A,B,E,F RWCU-TS-81 C,D,G,H	Once/Month (1)	Once/Operating Cycle	None

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)
Main Control Room Ventilation Isolation	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		
Main Control Room Vent Isolation		Once/6 Months		
Mechanical Vacuum Pump Isolation		Once/Operating Cycle		

NOTES FOR TABLES 4.2.A THROUGH 4.2.F

1. Initially once every month until exposure (M as defined on Figure 4.1.1) is 2.0 X 10^5; thereafter, according to Figure 4.1.1 (after NRC approval). The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of CNS.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of applying simulated inputs. Local alarm lights representing upscale and downscale trips will be verified but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level and high drywell pressure are not included on Table 4.2.A since they are tested on Table 4.1.2.

6. The logic system functional tests shall include an actuation of time delay relays and timers necessary for proper functioning of the trip systems.

7. These units are tested as part of the Core Spray System tests.

8. The flow bias comparator will be tested by putting one flow unit in "Test" (producing a rod block) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verifying that it will produce a rod block during the operating cycle.

9. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

10. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

11. Surveillance requirements for this system are defined in Table 4.21.A.1.

12. Surveillance requirements for this system are defined in Table 4.21.A.2.

13. This instrumentation is exempted from the instrument channel test definition. The instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels to test the alarm and trip functions.

14. Calibration shall be performed using a standard current source. The current source provides instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Paragraph VI.5.3.1 USAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 6 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case of accident, main steam line break outside the drywell, a trip setting of 150% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel clad temperatures peak at approximately 1000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section XIV.6.5 USAR.

Temperature monitoring instrumentation is provided in the main steam tunnel and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200°F for the main steam leak detection system. For large breaks, the high steam flow instrumentation is a backup to the temp. instrumentation.

High radiation monitors in the main steam tunnel have been provided to detect gross fuel failure as in the control rod drop accident. These monitors alert control room operators to potential fuel degradation by means of an alarm set at ≤1.5 times the normal background, and initiate a Group 7 isolation at ≤3 times the normal background.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below Specification 2.1.A.6. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in Section XIV.5 of the USAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN mode is not required.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that core uncovery is prevented and fission product release is within limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated May 4, 1992 (Reference 1), as supplemented by letters dated October 15, 1992 (Reference 2), January 13, 1993 (Reference 4), February 12, 1993 (Reference 8), and February 24, 1993 (Reference 9), Nebraska Public Power District (the District, NPPD, or the licensee) submitted a request for an amendment authorizing the removal of the Main Steam Line Radiation Monitor (MSLRM) reactor scram and Group I Containment Isolation functions, and modifying the Technical Specifications (TS) for Cooper Nuclear Station (CNS) accordingly. Specifically, this amendment will make the following changes to the TS.

(1) References to Main Steam Line Radiation Monitor RMP-RM-251 A, B, C, and D are removed from TS Table 3.1.1, "Reactor Protection System Instrumentation Requirements," TS Table 4.1.1, "Reactor Protection System (Scram Instrumentation) Functional Tests, Minimum Functional Test Frequencies For Safety Instrumentation and Control Circuits," and TS Table 4.1.2, "Reactor Protection System (Scram) Instrument Calibration Minimum Calibration Frequencies For Reactor Protection Instrument Channels."

(2) Action Statement "D" is removed from TS Table 3.1.1.

(3) Note 4 is deleted from TS Table 4.1.1.

(4) Note 3 is deleted from TS Table 4.1.2.

(5) The discussion in TS Bases 3.1 relating to reactor scram on high MSLRM signal is deleted.

(6) A new Action Statement "E" is added to TS Table 3.2.A, "Primary Containment and Reactor Vessel Isolation Instrumentation," to require isolation of the Reactor Water Sample Valves (Group 7) if the MSLRM becomes inoperable. Also, MSLRM is removed from the Group 1 Containment Isolation signal list in Table 3.2.A.

(7) Minor editorial changes are made to Action Statement "E" of TS Table 3.2.D, "Radiation Monitoring Systems That Initiate And/Or Isolate Systems."

(8) Surveillance requirements of the MSLRM are added to TS Table 4.2.A, "Primary Containment and Reactor Vessel Isolation System Test and Calibration Frequencies."

(9) References in TS Table 4.2.D, "Minimum Test and Calibration Frequencies For Radiation Monitoring Systems," to surveillance requirements associated with the Mechanical Vacuum Pump isolation (provided by the MSLRM) in TS Tables 4.1.1 and 4.1.2 are changed to TS Table 4.2.A to reflect the relocation of MSLRM surveillance requirements from Tables 4.1.1 and 4.1.2 to Table 4.2.D.

(10) Note 5 to TS Tables 4.2.A through 4.2.F is revised to delete reference to the MSLRM. New notes 13 and 14 have been added to address MSLRM surveillance requirements.

(11) TS Bases 3.2 has been revised to reflect the removal of the Main Steam Isolation Valve (MSIV) closure function from the MSLRM.

The plant modifications authorized by this amendment will provide a number of operational benefits, while improving radiological release management associated with the Control Rod Drop Accident (CRDA). The October 15, 1992, and January 13, February 12, and February 24, 1993, NPPD letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination published in the Federal Register on July 8, 1992, at 57 FR 30252.

2.0 DISCUSSION

Boiling Water Reactors (BWRs) are equipped with radiation monitors which are located on the main steam lines downstream from the main steam isolation valves (MSIVs). The MSLRMs detect moderate-to-large fuel failures and close the MSIVs to stop the release of radioactivity into the steam lines. On detection of high radiation in the main steam line, trip circuits automatically shut down the reactor and close the MSIVs.

On July 9, 1987 (Reference 5), the Boiling Water Reactor Owner's Group (BWROG) requested that the NRC staff review NEDO-31400 (Reference 6), a topical report prepared by the General Electric Company, which evaluated the elimination of the MSIV closure function and reactor scram function of the MSLRM. NEDO-31400 showed that the radiological release consequences of the bounding accident (the CRDA) are within the NRC staff's acceptance criteria even without the automatic MSIV closure and reactor trip. The NRC staff reviewed NEDO-31400 and concluded that the proposed changes were acceptable. However, the staff's generic approval of NEDO-31400 imposed three additional requirements which must be met by licensees desiring to implement the proposed changes for specific plants. The NRC staff's safety evaluation of NEDO-31400 and the additional staff requirements were provided in Reference 7.

The additional requirements that must be met by licensees are:

(1) The licensee must demonstrate that the assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

(2) The licensee must include sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.

(3) The licensee must standardize the MSLRM and offgas radiation monitor alarm setpoints at 1.5 times the nominal nitrogen-16 background dose rate at the monitor locations, and commit to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors, or both, exceed their alarm setpoints.

3.0 EVALUATION

The CNS licensee has proposed several changes to the TS to implement the deletion of the MSLRM containment isolation and reactor trip functions, and to meet the three additional requirements imposed by Reference 7.

With regard to requirement (1) of Reference 7, the licensee provided, in Reference 4 (as corrected by Reference 9), a table comparing the key input assumptions used in NEDO-31400 and those used for the CNS analysis. The staff has reviewed the table and finds that it provides an acceptable demonstration that the CNS CRDA analysis is bounded by the assumptions used in the generic analysis of NEDO-31400. On this basis the NRC staff concludes that requirement (1) of Reference 7 has been met.

With regard to requirement (2) of Reference 7, the licensee provided, in Reference 4, a discussion of the existing procedures and procedural changes that will be made to assure that expeditious actions will be taken to minimize both occupational exposure and environmental releases during periods when increased coolant activity is experienced. The NRC staff has reviewed the licensee's discussion of plant procedures and procedural changes and concludes that it is acceptable. On this basis the staff finds that requirement (2) of Reference 7 has been met.

With regard to requirement (3) of Reference 7, the licensee stated in Reference 4 that the MSLRM high alarm setpoint will be retained at 1.5 times background, and the offgas radiation monitor high alarm will be set at slightly greater than 1.5 times background. The offgas radiation monitor alarm will be set slightly above 1.5 times background in order to avoid nuisance alarms in the control room while performing weekly grab sampling or monthly source checks. In Reference 4, the licensee provides a description of the operator actions to be taken following high and high-high alarms from the

MSLRM or offgas radiation monitor. The actions include prompt sampling of the reactor coolant following a high alarm. The NRC staff has reviewed the licensee's commitments regarding standardizing the MSLRM and offgas radiation monitor alarm setpoints and prompt sampling of reactor coolant on an alarm and finds them acceptable. On this basis the staff concludes that requirement (3) of Reference 7 has been adequately met.

In summary, the NRC staff has reviewed the proposed TS changes to implement removal of the MSLRM Reactor Scram and Group 1 Containment Isolation closure functions. The NRC staff has previously reviewed and found acceptable the generic topical report prepared by the General Electric Company for implementing these changes. The staff safety evaluation is documented in Reference 7. The staff approval of these changes for specific plants is conditional on the licensee meeting the three additional requirements defined in Reference 7. The staff review of the proposed changes to the CNS TS concludes that the licensee's application and the proposed changes meet the three additional requirements. Accordingly, the proposed TS changes discussed in References 1, 2 and 4 are acceptable to the staff and are hereby approved.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 30252). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

(1) Letter and enclosures from G. R. Horn, Nebraska Public Power District, to USNRC, dated May 4, 1992, "Proposed Change No. 100 to Technical Specifications, Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

(2) Letter and enclosures from G. R. Horn, Nebraska Public Power District, to USNRC, dated October 15, 1992, "Revision to Proposed Change No. 100 to Technical Specifications, Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

(3) Letter from H. Rood, NRC, to G. R. Horn, Nebraska Public Power District, dated December 1, 1992, "Request for Additional Information (RAI) Related to Proposed Change No. 100 to the Cooper Nuclear Station Technical Specifications, Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions (TAC No. 83768)."

(4) Letter from G. R. Horn, Nebraska Public Power District, to USNRC, dated January 13, 1993, "Response to Request for Additional Information Related to Proposed Change No. 100 to Technical Specifications, 'Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions,' (TAC No. M83768), Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

(5) Letter and enclosures from R. Janeck, BWR Owner's Group, to J. Funches, NRC, dated July 9, 1987, requesting an NRC review of the generic Topical Report NEDO-31400.

(6) NEDO-31400, "Safety Evaluation of Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," the General Electric Company, May 1987.

(7) Letter and enclosures from A. Thadani, NRC, to George J. Beck, Chairman, BWR Owner's Group, dated May 15, 1991, "Acceptance for Referencing of Licensing Topical Report NEDO-31400, 'Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of Main Steam Line Radiation Monitor.'"

(8) Letter and enclosures from G. R. Horn, Nebraska Public Power District, to USNRC, dated February 12, 1993, "Submittal of Pages Inadvertently Omitted from Proposed Change No. 100 to Technical Specifications, 'Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions,' Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

(9) Letter from G. R. Horn, Nebraska Public Power District, to USNRC, dated February 24, 1993, "Correction to Response to Request for Additional Information Regarding Proposed Change No. 100 to Technical Specifications Elimination of Main Steam Line Radiation Monitor Scram and Isolation Functions, Cooper Nuclear Station, NRC Docket No. 50-298, DPR-46."

Principal Contributor: H. Rood

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