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Docket No. 50-298

OCT 15 1974

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Manager
Licensing and Quality Assurance
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment includes Change No. 7 to the Technical Specifications and is in response to part of your request dated May 28, 1974.

The Safety Evaluation and a copy of the Federal Register Notice relating to this action are also enclosed.

Sincerely,

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 4 to Facility Operating License No. DPR-46
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See attached

bcc: H. J. McAlduff, ORO
J. R. Buchanan, ORNL
T. B. Abernathy, DTIE

LB

OFFICE ▶	L:ORB-2	L:ORB-2	L:ORB-2	OGC	L:AD/ORs	
SURNAME ▶	x7403 JSapir:esp	RMDiggs	DLZiemann	R. Kinsey	KRG KRGoller	
DATE ▶	8/ /74	8/27/74	8/23/74	10/8/74	10/11/74	

10/2/74 10/2/74 10/2/74

Form AEC-318 (Rev. 9-53) AECM 0240 GPO 16-81465-1 445-678

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Auburn Public Library
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Lincoln, Nebraska 68509

Mr. Ed Vest
Environmental Protection Agency
1735 Baltimore Avenue
Kansas City, Missouri 64108

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DATE ▶						

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-46

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for amendment by Nebraska Public Power District (the licensee) dated May 28, 1974, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 7."

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

FOR THE ATOMIC ENERGY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 7 to the
Technical Specifications

Date of Issuance. 1974

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DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 7 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

The Technical Specifications contained in Appendix A, attached to Facility Operating License No. DPR-46, are hereby changed as follows and revised pages incorporating these changes are appended hereto:

1. On page 3, Definition L, item 3; delete items a and b and place these items on page 5, under Definition W as items 1. and 2.

2. On page 163, Specification 3.7.A.3.b, change "locked closed" to "secured in the closed position".

3. On page 32, Table 4.1.1

   IRM High Flux and Inoperative - change column 4, Minimum Frequency to read: "Before each startup and weekly when required to be operable".

   APRM High Flux (15%); - change column 2, Group, to read: "C".

   On page 34, add the following sentence to Note 3:

   "If reactor startups occur more frequently than once per week, the maximum functional test frequency need not exceed once per week".

4. On page 107, Specification 4.4.A.1, remove the words "functionally tested" and replace with "tested for operability".

5. On page 107, Specification 4.4.A.2.b, replace the first 2 sentences with "Manually initiate the system, except explosive valves, and pump boron solution from the Standby Liquid Control System through the recirculation path".

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6. On page 57, column 3, change the Setting Limit for HPCI-PS-2787-H from ≤95 psig to ≥85 psig.

7. Change the "<" to "≤" on the following time delays:

   a. Page 53, CS-TDR-K16 A&B

   b. Page 57, HPCI-TDR-K14

8. Change the following Setting Limits (Column 3) of Table 3.2.B:

   a. Page 53, CS-TDR-K16 from 9.5<T<10.5 seconds to 9≤T≤11 seconds.

   b. Page 55, RHR-TDR-K81 A&B, from 10 min.±5 min. to 9≤T≤11 min.

   c. Page 55, RHR-TDR-K86 A&B, from 10 min.±5 min. to 9≤T≤11 min.

   d. Page 55, RHR-TDR-K45 1A&1B, from 5 min.±5 min. to 4.25≤T≤5.75 min.

   e. Page 55, RHR-TDR-K93 A&B, from 2 min.±5 sec. to 1.8≤T≤2.2 min.

   f. Page 57, HPCI-TDR-K14, from 14<T<16 sec. to 13.5≤T≤16.5 sec.

   g. Page 58, RCIC-TDR-K9, from 14<T<16 sec. to 13.5≤T≤16.5 sec.

9. a. Page 61, Table 3.2.C. Add the following information to the bottom of the table:

   Function: RSCS Rod Group C Bypass
   Trip Level Setting: ≥20% Core Thermal Power
   Minimum Number of Operable Instrument Channels: (11)

   b. Page 62, Notes for Table 3.2.C. Add the following Note:

   "11. There are two pressure transmitters, which provide the signal to allow Group C rod withdrawal. Failure of either unit will not allow the Group C withdrawal. These units are required to function only until 6500 MWD/T at which time, the Group C interlock will be removed after AEC approval.

   As described in the Bases for Specification 3.3.B.3, the RSCS operation is required below 20% core thermal power. Therefore, the pressure switches will be calibrated to 22% core thermal power to allow for the instrumentation and calculation accuracy".

c. Page 77, Table 4.2.C. Add the following information to the bottom of the table.

Function. RSCS Rod Group C Bypass
Functional Test Frequency. (1)(11)
Calibration Frequency. Once/3 Months
Instrument Check. N. A.

d. Page 81, Notes for Table 4.2.A through 4.2.F. Add the following Note:

11. The RSCS Rod Group C Bypass function is required for the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

e. Page 86, add the following paragraph between the 4th and 5th paragraphs.

The RSCS Rod Group C Bypass function is required only during the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

10. On page 164, Specification 4.7.A.4.d change "3.4.A.4.c" to "3.7.A.4.c".

11. Pages 28 and 29 correct minor errors on Note references and clarify other items. Replace these pages with the attached revised pages.

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8. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

9. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

J. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

L. Mode - The reactor mode is established by the mode selector-switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor
7 mode switch is in the shutdown mode position.

4. Startup/Hot Standby - In this mode the reactor protection scram trips

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" or "Refuel" position.

7 1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F and the reactor vessel vented.

X. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

Y. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable or as practicable following its return to service.

Z. Thermal Parameters

1. Minimum Critical Heat Flux Ratio (MCHFR) - The lowest in-core ratio of critical heat flux (that heat flux which results in transition boiling) to the actual heat flux at the same location.

2. Peaking Factor - The ratio of the maximum fuel rod surface heat flux in any assembly to the average surface heat flux of the core.

3. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3.7.A (cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc.

b. From and after the date that one of
the pressure suppression chamber-
reactor building vacuum breakers is
made or found to be inoperable for
7 any reason, the vacuum breaker shall
be secured in the closed position and
reactor operation is permissible only
during the succeeding seven days
unless such vacuum breaker is sooner
made operable, provided that the repair
procedure does not violate primary
containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below.

b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met.

c. The total leakage between the drywell and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice.

4.7.A (cont'd.)

*torus corrosion or leakage.

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months.

b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.

b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

COOPER NUCLEAR STATION
TABLE 4.1.1
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)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Once/3 Months
RPS Channel Test Switch (5)	A	Trip Channel and Alarm	Each Refueling Outage
IRM			
High Flux	C	Trip Channel and Alarm (4)	Before each startup and weekly when required to be operable 7
Inoperative	C	Trip Channel and Alarm	Before each startup and weekly when required to be operable 7
APRM			
7 High Flux (15%)	C	Trip Output Relays (4)	Before Each Startup and Weekly when required to be operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	Trip Output Relays	Once/Month (1)
High Reactor Pressure NBI-PS-55 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure PC-PS-12 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water level (6) NBI-LIS-101 A,B,C, & D	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 AEC. 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
7 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. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

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.

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shutdown the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B. below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System shall be shown by the performance of the following tests:

1. At least once per month each pump loop shall be tested for operability by recirculating demineralized water to the test tank. 7

2. At least once during each operating cycle:

a. Check that the settings of the system relief valves are $1400 < P < 1680$ psig and the valves will reset at $P \geq 1215$ psig.

b. Manually initiate the system, except explosive valves, and pump boron solution from the Standby Liquid Control System through the recirculation path. Minimum pump flow rate of 38.2 gpm against a system head of 1215 psig shall be verified. After pumping boron solution the system will be flushed with demineralized water. 7

c. Manually initiate one of the Standby Liquid Control System Pumps and

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TABLE 3.2.B (Page 1)

CIRCUITRY REQUIREMENTS CORE SPRAY SYSTEM

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System	Action Required When Component Operability Is Not Assured (1)
Reactor Low Water Level	NBI-LIS-72 A,B,C, & D	$\geq$-145.5 of Indicated Level	2	A
Reactor Low Pressure	NBI-PS-52 A & C NBI-PIS-52 B & D	$\leq$450 psig	2	A
Drywell High Pressure	PC-PS-101, A,B,C, & D	$\leq$2 psig	2	A
Core Spray Pump Disch. Press.	CS-PS-44, A & B CS-PS-37, A & B	$\leq$165 psig.	2	A
7 Core Spray Pump Time Delay	CS-TDR-K16 A & B	$9 \leq T \leq 11$ seconds	1	B
Core Spray Minimum Flow	CS-AL - 45, A & B	$\leq$1200 gpm	1(3)	A
Low Voltage Relay Emerg. Bus	27X1 - 1F & 1G 27X2 - 1F & 1G	Loss of Voltage	1	B
Aux. Bus Low Voltage Relay	27X3 - 1A & 1B	Loss of Voltage	1	B
Pump Discharge Line Low Pressure	CM-PS-73, A & B	$\geq$10 psig	(3)	D

COOPER NUCLEAR STATION
TABLE 3.2.B (Page 3)
RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
RHR Pump Low Flow	RHR-dPIS-125 A & B	≥2500 gpm	1	A
Break Detection Time Delays	RHR-TDR-K28, A & B	0.25≤T≤0.75 sec.	1	A
	RHR-TDR-K40, A & B	0.25≤T<0.75 sec.	1	A
	RHR-TDR-K34, A & B	1.5<T<2.5 sec.	1	A
	(RHR-TDR-K81, A & B)	9≤T≤11 min.	1	A
	RHR-TDR-K86, A & B	9≤T≤11 min.	1	A
	RHR-TDR-K45, 1A & 1B	4.25≤T≤5.75 min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75, A & B	4.5≤T≤5.5 sec.	1	A
	RHR-TDR-K70, A & B	≤.5 sec.	1	A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	1.8≤T≤2.2 min.	1	B
RHR Crosstie Valve Position	RHR-LMS-2	N.A.	(3)	D
Bus 1A Low Volt. Aux. Relay	27 X 3/1A	Loss of Voltage	1	B
Bus 1B Low Volt. Aux. Relay	27 X 3/1B	Loss of Voltage	1	B
Bus 1F Low Volt. Aux. Relays	27 X 1/1F	Loss of Voltage	1	B
	27 X 2/1F	Loss of Voltage	1	B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G	Loss of Voltage	1	B
	27 X 2/1G	Loss of Voltage		
Pump Discharge Line Low Pressure	CM-PS-266	≥5 psig	(3)	D

COOPER NUCLEAR STATION
TABLE 3.2.B (Page 5)
HPCI SYSTEM CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
Suppression Chamber High Water Level	HPCI-LS-91 A & B	2½" H_2O (5" Above Normal)	1(2)	A
HPCI Gland Seal Cond. Hotwell Level	HPCI-LS-356 B HPCI-LS-356 A	≥18" ≤46"	1(3) 1(3)	A A
HPCI Turbine Stop Valve Monitor	HPCI-LMS-4	N.A.	1(2)	B
Suppression Chamber HPCI Suction Valve 23-58	HPCI-LMS-2	N.A.	1(2)	A
HPCI Control Oil Pressure Low	HPCI-PS-2787-H HPCI-PS-2787-L	≥85 psig ≥20 psig	1(2)	B
Turbine Conditional Supervisory Alarm Actuation Timer	HPCI-TDR-K14	13.5≤T≤16.5 sec.	1(3)	E
Pump Discharge Line Low Pressure	CM-PS-268	≥10 psig	(3)	D

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TABLE 3.2.B (Page 6)
REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	≤25 psig	1(2)	A
RCIC Low Pump Suction Press.	RCIC-PS-67-1	≥-15" Hg	1(2)	(A)
RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D RCIC-TS-80, A,B,C, & D RCIC-TS-81, A,B,C, & D RCIC-TS-82, A,B,C, & D	≤200°F	2(4)	A
RCIC Steam Line High ΔP	RCIC-dPIS-83 & 84	≤+180" H_2O	1	A
RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C & D	≥50 psig	2(2)	A
RCIC Low Pump Disch. Flow	RCIC-FIS-57	≥40 gpm	1(2)	A
Pump Discharge Line Low Pressure	CM-PS-269	≥10 psig	(3)	D
RCIC Turbine Conditional Supervisory Alarm Timer	RCIC-TDR-K9	13.5≤T≤16.5 sec.	(3)	E
Reactor Low Water Level	10A-K80, A & B 10A-K79, A & B (NBI-LIS-72, A,B,C, & D)	≥-37" Indicated Level	2(2)	A
Reactor High Water Level	NBI-LIS-101, A & C #3	≤+58.5 Indicated Level	2(2)	A

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System (5)
APRM Upscale (Flow Bias)	≤(0.66W + 42) (2)	2(1)
APRM Upscale (Startup)	≤12%	2(1)
APRM Downscale (9)	≥ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	≤(0.66W + 41) (2)	1
RBM Downscale (9)	≥ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	≤ 108/125 of Full Scale	3(1)
IRM Downscale (3) (8)	≥ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	≤ 1×10^5 Counts/Second	1(1) (6)
SRM Detector Not Full In (4) (8)	(≥ 100 cps)	1(1) (6)
SRM Inoperative (8)	(10a)	1(1) (6)
Flow Bias Comparator	≤ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	≤ 110% Recirc. Flow	1
SRM Downscale (8) (7)	≥ 3 Counts/Second	1(1) (6)
RSCS Rod Group C Bypass	≥20% Core Thermal Power	(11)

7

11. There are two pressure transmitters, which provide the signal to allow Group C rod withdrawal. Failure of either unit will not allow the Group C withdrawal. These units are required to function only until 6500 MWD/T at which time, the Group C interlock will be removed after AEC approval.

As described on page 101, the RSCS operation is required below 20% core thermal power. Therefore, the pressure switches will be calibrated to 22% core thermal power to allow for the instrumentation and calculation accuracy.

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TABLE 4.2.C

SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Flow Bias)	(1) (3)	Once/6 Months	Once/Day
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/Oper. Cycle	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
RSCS Rod Group C Bypass	(1) (11)	Once/3 Months	N.A.

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 2.0×10^5; thereafter, according to Figure 4.1.1(after AEC 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, high drywell pressure and high radiation main steam line tunnel 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 1/2 scram) 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. The RSCS Rod Group C Bypass function is required for the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

7 The RSCS Rod Group C Bypass function is required only during the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

The refueling interlocks also operate one logic channel, and are required for a safety only when the mode switch is in the refueling position.

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

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the stack release rate limit given in Specification 2.4.3.a of the Environmental Technical Specifications is not exceeded.

Two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. The trip is actuated by one hi-hi or two downscale indications.

LIMITING CONDITIONS FOR OPERATION

3.7.A (cont'd.)

d. If specifications 3.7.A.4.a,b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours.

5. Oxygen Concentration

a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons.

d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.2 below, both trains of the standby gas treatment system and the diesel generators

SURVEILLANCE REQUIREMENTS

4.7.A (cont'd.)

c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation of the position indication switch. The differential pressure setpoint shall be verified.

d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met. 7

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect.

B. Standby Gas Treatment System

1.a. At least once per operating cycle it shall be demonstrated that pressures drop across the combined high efficiency

COOPER NUCLEAR STATION
TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position				Trip Level Setting	Minimum Number of Operable Channels Per Trip System (1)	Action Required When Equipment Operability is Not Assured (1)
	Shutdown	Startup	Refuel	Run			
Mode Switch in shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	$\le$120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	$\le (0.66W+54)\frac{2.61}{P.F.}$ (14)	2	A or C
High Flux	X(7)	X (9)	X (9)	(16)	$\le$15% Rated Power		A or C
Inoperative		X (9)	X (9)	X	(13)	2	A or C
Downscale		(11)		X(12)	$\ge$ 2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X (9)	X(10)	X	$\le$ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X (9)	X(8)	X	$\le$2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	$\ge$ + 12.5 in. indicated level	2	A or D
Scram Discharge Volume High Water Level CRD-LS-231 A,B,C, & D	X (2)(7)	X	X (2)	X	$\le$36 gallons	2	A

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

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position				Trip Level Setting	Minimum Number of Operable Channels Per Trip System (1)	Action Required When Equipment Operability is Not Assured (1)
	Shutdown	Startup	Refuel	Run			
Main Steam Line High Radiation RMP-RM-251, A,B,C, & D		X(9)		X	≤ 6 Times normal full power back ground	2	A or D
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D		X(3)(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 psi 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
Reactor Pressure Permissive NBI-PS-51 A,B,C, & D		X(9)		X	≤1000 psig	2	A or C
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	≤ 30% first stage press.	2	A or B

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NO. DPR-46

CHANGE NO. 7 TO THE TECHNICAL SPECIFICATIONS

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

Introduction

By letter dated May 28, 1974, Nebraska Public Power District (NPPD) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS).

Discussion

The propesed changes transmitted in Attachment 1 to the May 28, 1974 submittal have been issued, as modified, in Amendment No. 2 incorporating Change No. 5, dated August 15, 1974.

This safety evaluation addresses the changes transmitted in Attachment 2 to the May 28, 1974 submittal, except for the following.

(1) The eleventh proposed change pertaining to the operability of the core spray pump minimum flow valve and the nineteenth proposed change pertaining to primary containment surveillance instrumentation require additional information from the licensee. These changes will be addressed at a later date.

(2) Actioneon proposed changes fifteen, sixteen, and seventeen is being deferred pending further review.

(3) The twentieth change concerning fuel densification is being evaluated in a separate review.

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Proposed change #1 would transfer the definition of hot and cold shutdown to a more appropriate part of Section 1.0 (Definitions) of the Technical Specifications. This change clarifies the affected definitions and is acceptable. This change is implemented by revising Definition L., "Mode" on page 3 and Definition W., "Shutdown" on page 5.

Proposed changes #2 and #3 would remove requirements to perform instrument checks on several of the Reactor Core Isolation Cooling (RCIC) system and the Rod Withdrawal Block (RWB) instrumentation channels. Although we agree with NPPD that the definition of "instrument check" as defined in Definition I.4 does not pertain to relays, the intent of the surveillance requirements is to perform a check on the instrument channels incorporating these relays. We have concluded that the proposed change is therefore not acceptable and should be denied.

Proposed change #4 would permit procedural controls to replace physical locks to assure that inoperable suppression chamber-to-reactor building vacuum breakers remain in a closed position. The purpose of these vacuum breakers is to equalize pressure between the suppression chamber and the reactor building. To assure that primary containment integrity is maintained, these vacuum breakers must be in the closed condition during normal operation. Since there are two 100 percent capacity vacuum breaker systems, the Technical Specifications allow reactor operation for seven days with one inoperable vacuum breaker in the closed condition. The specific wording of Specification 3.7.A.3.b required the inoperable vacuum breaker to be "locked closed". Since there is no locking mechanism on these breakers, the requirement cannot be literally met and NPPD requested a change that would allow procedural controls. Considering the imposed seven day time limit, we have concluded that procedural controls, such as appropriate tags, will adequately assure that the vacuum breaker will not be inadvertently manually opened. We have, however, modified the proposed wording to use the standard phrase "secured in the closed position". We have concluded that the proposed change as modified, does not reduce the safety provisions of the Technical Specifications and is therefore acceptable. This change is implemented by revising Specification 3.7.A.3.b on page 163 of the Technical Specifications.

Proposed change #5 would limit the functional test requirements on Intermediate Range Monitor (IRM) and Average Power Monitor (APRM) instrument channels to no more than once per week. The original specifications (Table 4.1.1) require functional tests on the IRM and APRM 15% channels before each startup. This specification results in a high frequency of functional tests during the startup program. Many years of experience in many nuclear facilities have demonstrated the high reliability and

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satisfactory performance of these channels. We have concluded that the proposed change does not significantly reduce the safety provisions in the Technical Specifications and is therefore acceptable. In conjunction with this change, we have incorporated several additional changes into Table 4.1.1 to correct a typographical error and clarify the original intent. These changes are incorporated into Table 4.1.1 on page 32 and the corresponding notes on page 34.

Proposed change #6 would limit the functional test requirements on the Rod Sequence Control System (RSCS) and Rod Worth Minimizer (RWM) to no more than once per week. Those systems prevent the withdrawal of an out-of-sequence control rod and are required to be operable whenever rod reactivity worths are such that a single error Rod Drop Accident could result in unacceptable energy releases. Current specifications require these systems to be tested prior to entering a condition when they are required to be operable; namely, prior to the start of control rod withdrawal towards criticality and prior to attaining 20% of rated power during shutdown.

The RSCS and RWM are relatively new and complex systems. The staff has concluded that sufficient experience with and reliability of these systems has not been established to justify relaxation of the functional test requirements. The proposed change is therefore unacceptable and is therefore denied.

Proposed change #7 clarifies the intent of the surveillance requirements on the Standby Liquid Control System (SLC) (Specification 4.4.A). A functional test of the SLC is required by Specification 4.4.A.2.b to be performed once per operating cycle. The intent of Specification 4.4.A.1 is to test the SLC pumps for operability. The proposed change incorporates this clarification, does not reduce the safety provisions of the Technical Specifications and is therefore acceptable. This change is implemented by revising Specification 4.4.A.1 on page 107 of the Technical Specifications.

Proposed change #8 would remove the requirement of pumping boron solution back into the solution tank during the SLC functional test, and would thereby permit pumping into the test tank. The objective of this test is to verify that the specified pump flow rate is obtained from the solution tank. This objective can be met more easily and more accurately by observing the flow into the test tank than by taking suction from and' discharging to the solution tank. We have concluded that the proposed change does not reduce the safety provisions of the Technical Specifications and is therefore acceptable. This change is implemented by revising Specification 4.4.A.2.b.

Proposed change #9 would correct an error in the setpoint specification for the High Pressure Coolant Injection (HPCI) control oil pressure switch (Table 3.2.B). The HPCI initiation signal automatically starts an auxiliary oil pump which provides hydraulic pressure during the HPCI turbine startup. When the turbine attains sufficient speed, the required hydraulic pressure is supplied by a shaft driven oil pump and the auxiliary oil pump is shut off by the control oil pressure switch. It is important that the auxiliary oil pump operates whenever the shaft driven oil pump does not develop the required pressure (~80 psig). Failure of the auxiliary oil pump to shut off does not constitute a safety problem. Whereas the current set point specification of ≤95 psig allows a low setting which could result in a premature pump trip, the proposed setpoint of ≥85 psig assures continued operation of the auxiliary oil pump until sufficient pressure is available. The proposed setpoint is therefore more appropriate, increases plant safety and is therefore acceptable.

Proposed change #10 would change the symbol from "<" to "≤" in the specification of time delays in the Core Standby Cooling Systems. This change has no significant effect on the actual limits. We have concluded the proposed change does not modify the safety provisions of the Technical Specifications and is therefore acceptable. This change is implemented by an appropriate change on page 57 of the Technical Specifications.

Proposed change #12 slightly relaxes the range on time delay relay setpoints in the Core Spray, Residual Heat Removal, High Pressure, Coolant Injection and Reactor Core Isolation Coolant systems to reflect instrument design accuracies. The first relay delays the automatic start of the core spray pump if offsite power is lost so as not to overload the emergency diesel generator. The increased setpoint range will not have a significant effect on either the diesel generator loads or the peak clad temperature. The next four delay relays prevent the operator from assuming control of various LPCI components after automatic initiation, until the system has had sufficient time to carry out its design function. The new set point range will not prevent proper operation of the LPCI system or interfere with subsequently required operator actions. The final two delay relays, delay annunciation of low flow during HPCI and RCIC startup.

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The increased setpoint range will have no adverse effect upon satisfactory operation of these systems. We have concluded that the proposed changes do not modify the safety provisions of the Technical Specifications and are therefore acceptable. This change is implemented by appropriate revisions on pages 53, 55 and 57 of the Technical Specifications.

Proposed change #13 adds to the Technical Specifications instrumentation requirements which prevent operation of the RSCS Rod Group C Bypass below 20% of rated power. This instrumentation was approved by the staff in Change No. 2 to License No. DPR-46, dated April 17, 1974, and is currently in operation. The proposed change incorporates the channels into the Technical Specifications along with appropriate surveillance requirements and supporting bases. We have concluded that the proposed change increases the safety provisions of the Technical Specifications and is therefore acceptable. This change is implemented by additions to pages 61, 62, 77, 81 and 86 of the Technical Specifications.

Proposed change #14 corrects a typographical error on page 164, Specification 4.7.A.4.d. by changing the reference to Specification 3.4.A.4.C to read 3.7.A.4.C.

Proposed change #18 would modify Table 3.1.1, Reactor Protection System Instrumentation Requirements, on pages 28 and 29 of the Technical Specifications. We have reviewed these proposed changes and concluded they are essentially clarifications and corrections and do not modify the safety provisions of the Technical Specifications and are therefore acceptable. This change is implemented by replacing pages 28 and 29 with revised pages 28 and 29.

Conclusion

We have concluded, based on the reasons discussed above, that because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration. We also conclude that there is

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reasonable accurance (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 and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

/S/

Joseph L. Sapir
Operating Reactors Branch #2
Directorate of Licensing

/S/

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: OCT 15 1974

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. DPR-46 issued to Nebraska Public Power District which revised Technical Specifications for operation of the Cooper Nuclear Station, located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

This amendment changes and/or clarifies the provisions in the Technical Specifications relating to certain definitions, interlocks, surveillance requirements, setpoints and limiting conditions of operation to incorporate the results of experience gained during the course of the startup test program and the obtaining of system test data.

The application for the amendment complies with the standards and requirements of the Act and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

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For further details with respect to this action, see (1) the application for amendment dated May 28, 1974, (2) Amendment No. 4 to License No. DPR-46, with Change No. 7, and (3) the Commission's concurrently issued Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Auburn Public Library, 1118 - 15th Street, Auburn, Nebraska 68305. A copy of items (2) and (3) may be obtained upon request addressed to the United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 15th day of October, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

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Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43-16-81465-1 445-678

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 7 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

The Technical Specifications contained in Appendix A, attached to Facility Operating License No. DPR-46, are hereby changed as follows and revised pages incorporating these changes are appended hereto:

1. On page 3, Definition L, item 3; delete items a and b and place these items on page 5, under Definition W as items 1. and 2.

2. On page 163, Specification 3.7.A.3.b, change "locked closed" to "secured in the closed position".

3. On page 32, Table 4.1.1

   IRM High Flux and Inoperative - change column 4, Minimum Frequency to read: "Before each startup and weekly when required to be operable".

   APRM High Flux (15%); - change column 2, Group, to read: "C".

   On page 34, add the following sentence to Note 3:

   "If reactor startups occur more frequently than once per week, the maximum functional test frequency need not exceed once per week".

4. On page 107, Specification 4.4.A.1, remove the words "functionally tested" and replace with "tested for operability".

5. On page 107, Specification 4.4.A.2.b, replace the first 2 sentences with "Manually initiate the system, except explosive valves, and pump boron solution from the Standby Liquid Control System through the recirculation path".

6. On page 57, column 3, change the Setting Limit for HPCI-PS-2787-H from ≤ 95 psig to ≥ 85 psig.

7. Change the "$<$" to "$\leq$" on the following time delays:

   a. Page 53, CS-TDR-K16 A&B

   b. Page 57, HPCI-TDR-K14

8. Change the following Setting Limits (Column 3) of Table 3.2.B:

   a. Page 53, CS-TDR-K16 from $9.5<T<10.5$ seconds to $9 \leq T \leq 11$ seconds.

   b. Page 55, RHR-TDR-K81 A&B, from 10 min.± 5 min. to $9 \leq T \leq 11$ min.

   c. Page 55, RHR-TDR-K86 A&B, from 10 min.± 5 min. to $9 \leq T \leq 11$ min.

   d. Page 55, RHR-TDR-K45 1A&1B, from 5 min.± 5 min. to $4.25 \leq T \leq 5.75$ min.

   e. Page 55, RHR-TDR-K93 A&B, from 2 min.± 5 sec. to $1.8 \leq T \leq 2.2$ min.

   f. Page 57, HPCI-TDR-K14, from $14<T<16$ sec. to $13.5 \leq T \leq 16.5$ sec.

   g. Page 58, RCIC-TDR-K9, from $14<T<16$ sec. to $13.5 \leq T \leq 16.5$ sec.

9. a. Page 61, Table 3.2.C. Add the following information to the bottom of the table:

   Function: RSCS Rod Group C Bypass
   Trip Level Setting: $\geq 20\%$ Core Thermal Power
   Minimum Number of Operable Instrument Channels: (11)

   b. Page 62, Notes for Table 3.2.C. Add the following Note:

   "11. There are two pressure transmitters, which provide the signal to allow Group C rod withdrawal. Failure of either unit will not allow the Group C withdrawal. These units are required to function only until 6500 MWD/T at which time, the Group C interlock will be removed after AEC approval.

   As described in the Bases for Specification 3.3.B.3, the RSCS operation is required below 20% core thermal power. Therefore, the pressure switches will be calibrated to 22% core thermal power to allow for the instrumentation and calculation accuracy".

c. Page 77, Table 4.2.C. Add the following information to the bottom of the table.

Function: RSCS Rod Group C Bypass
Functional Test Frequency: (1)(11)
Calibration Frequency: Once/3 Months
Instrument Check: N. A.

d. Page 81, Notes for Table 4.2.A through 4.2.F. Add the following Note:

11. The RSCS Rod Group C Bypass function is required for the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

e. Page 86, add the following paragraph between the 4th and 5th paragraphs:

The RSCS Rod Group C Bypass function is required only during the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

10. On page 164, Specification 4.7.A.4.d change "3.4.A.4.c" to "3.7.A.4.c".

11. Pages 28 and 29 correct minor errors on Note references and clarify other items. Replace these pages with the attached revised pages.

8. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

9. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

J. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

L. Mode - The reactor mode is established by the mode selector-switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

7 3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position.

4. Startup/Hot Standby - In this mode the reactor protection scram trips

U. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

V. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

W. Shutdown - The reactor is in a shutdown condition when the mode switch is in the "Shutdown" or "Refuel" position.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

7 2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F and the reactor vessel vented.

X. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

Y. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable or as practicable following its return to service.

Z. Thermal Parameters

1. Minimum Critical Heat Flux Ratio (MCHFR) - The lowest in-core ratio of critical heat flux (that heat flux which results in transition boiling) to the actual heat flux at the same location.

2. Peaking Factor - The ratio of the maximum fuel rod surface heat flux in any assembly to the average surface heat flux of the core.

3. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3.7.A (cont'd.)

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for
7 any reason, the vacuum breaker shall be secured in the closed position and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below.

b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met.

c. The total leakage between the drywell and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice.

4.7.A (cont'd.)

torus corrosion or leakage.

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months.

b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.

b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

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TABLE 4.1.1
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)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Once/3 Months
RPS Channel Test Switch (5)	A	Trip Channel and Alarm	Each Refueling Outage
IRM			
High Flux	C	Trip Channel and Alarm (4)	Before each startup and weekly when required to be operable 7
Inoperative	C	Trip Channel and Alarm	Before each startup and weekly when required to be operable 7
APRM			
7 High Flux (15%)	C	Trip Output Relays (4)	Before Each Startup and Weekly when required to be operable
High Flux	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	Trip Output Relays	Once/Month (1)
High Reactor Pressure NBI-PS-55 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure PC-PS-12 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water level (6) NBI-LIS-101 A,B,C, & D	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 x 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 AEC. 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
7 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. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

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.

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shutdown the reactor and maintain the shutdown condition without the use of control rods.

Specification:

A. Normal System Availability

During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B. below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal System Availability

The operability of the Standby Liquid Control System shall be shown by the performance of the following tests:

1. At least once per month each pump loop shall be tested for operability by recirculating demineralized water to the test tank. 7

2. At least once during each operating cycle:

a. Check that the settings of the system relief valves are $1400 < P < 1680$ psig and the valves will reset at $P \geq 1215$ psig.

b. Manually initiate the system, except explosive valves, and pump boron solution from the Standby Liquid Control System through the recirculation path. Minimum pump flow rate of 38.2 gpm against a system head of 1215 psig shall be verified. After pumping boron solution the system will be flushed with demineralized water. 7 7

c. Manually initiate one of the Standby Liquid Control System Pumps and

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TABLE 3.2.B (Page 1)
CIRCUITRY REQUIREMENTS CORE SPRAY SYSTEM

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System	Action Required When Component Operability Is Not Assured (1)
Reactor Low Water Level	NBI-LIS-72 A,B,C, & D	$\geq$-145.5 of Indicated Level	2	A
Reactor Low Pressure	NBI-PS-52 A & C NBI-PIS-52 B & D	$\leq$450 psig	2	A
Drywell High Pressure	PC-PS-101, A,B,C, & D	$\leq$2 psig	2	A
Core Spray Pump Disch. Press.	CS-PS-44, A & B CS-PS-37, A & B	$\leq$165 psig.	2	A
7 Core Spray Pump Time Delay	CS-TDR-K16 A & B	$9 \leq T \leq 11$ seconds	1	B
Core Spray Minimum Flow	CS-AL - 45, A & B	$\leq$1200 gpm	1(3)	A
Low Voltage Relay Emerg. Bus	27X1 - 1F & 1G 27X2 - 1F & 1G	Loss of Voltage	1	B
Aux. Bus Low Voltage Relay	27X3 - 1A & 1B	Loss of Voltage	1	B
Pump Discharge Line Low Pressure	CM-PS-73, A & B	$\geq$10 psig	(3)	D

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TABLE 3.2.B (Page 3)

RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
RHR Pump Low Flow	RHR-dPIS-125 A & B	≥2500 gpm	1	A
Break Detection Time Delays	RHR-TDR-K28, A & B	$0.25 \leq T \leq 0.75$ sec.	1	A
	RHR-TDR-K40, A & B	$0.25 \leq T \leq 0.75$ sec.	1	A
	RHR-TDR-K34, A & B	$1.5 \leq T \leq 2.5$ sec.	1	A
	(RHR-TDR-K81, A & B)	$9 \leq T \leq 11$ min.	1	A
	RHR-TDR-K86, A & B	$9 \leq T \leq 11$ min.	1	A
	RHR-TDR-K45, 1A & 1B	$4.25 \leq T \leq 5.75$ min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75, A & B	$4.5 \leq T \leq 5.5$ sec.	1	A
	RHR-TDR-K70, A & B	≤.5 sec.	1	A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	$1.8 \leq T \leq 2.2$ min.	1	B
RHR Crosstie Valve Position	RHR-LMS-2	N.A.	(3)	D
Bus 1A Low Volt. Aux. Relay	27 X 3/1A	Loss of Voltage	1	B
Bus 1B Low Volt. Aux. Relay	27 X 3/1B	Loss of Voltage	1	B
Bus 1F Low Volt. Aux. Relays	27 X 1/1F	Loss of Voltage	1	B
	27 X 2/1F	Loss of Voltage	1	B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G	Loss of Voltage	1	B
	27 X 2/1G	Loss of Voltage		
Pump Discharge Line Low Pressure	CM-PS-266	≥5 psig	(3)	D

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TABLE 3.2.B (Page 5)
HPCI SYSTEM CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
Suppression Chamber High Water Level	HPCI-LS-91 A & B	2½" H_2O (5" Above Normal)	1(2)	A
HPCI Gland Seal Cond. Hotwell Level	HPCI-LS-356 B HPCI-LS-356 A	≥18" ≤46"	1(3) 1(3)	A A
HPCI Turbine Stop Valve Monitor	HPCI-LMS-4	N.A.	1(2)	B
Suppression Chamber HPCI Suction Valve 23-58	HPCI-LMS-2	N.A.	1(2)	A
7 HPCI Control Oil Pressure Low	HPCI-PS-2787-H HPCI-PS-2787-L	>85 psig ≥20 psig	1(2)	B
7 Turbine Conditional Supervisory Alarm Actuation Timer	HPCI-TDR-K14	13.5≤T≤16.5 sec.	1(3)	E
Pump Discharge Line Low Pressure	CM-PS-268	≥10 psig	(3)	D

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TABLE 3.2.B (Page 6)

REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	≤25 psig	1(2)	A
RCIC Low Pump Suction Press.	RCIC-PS-67-1	≤-15" Hg	1(2)	(A)
RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D RCIC-TS-80, A,B,C, & D RCIC-TS-81, A,B,C, & D RCIC-TS-82, A,B,C, & D	≤200°F	2(4)	A
RCIC Steam Line High ΔP	RCIC-dPIS-83 & 84	≤+180" H_2O	1	A
RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C & D	≥50 psig	2(2)	A
RCIC Low Pump Disch. Flow	RCIC-FIS-57	≥40 gpm	1(2)	A
Pump Discharge Line Low Pressure	CM-PS-269	≥10 psig	(3)	D
7 RCIC Turbine Condition-al Supervisory Alarm Timer	RCIC-TDR-K9	13.5≤T≤16.5 sec.	(3)	E
Reactor Low Water Level	10A-K80, A & B 10A-K79, A & B (NBI-LIS-72, A,B,C, & D)	≥-37" Indicated Level	2(2)	A
Reactor High Water Level	NBI-LIS-101, A & C #3	≤+58.5 Indicated Level	2(2)	A

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System (5)
APRM Upscale (Flow Bias)	≤(0.66W + 42) (2)	2(1)
APRM Upscale (Startup)	≤12%	2(1)
APRM Downscale (9)	≥ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	≤(0.66W + 41) (2)	1
RBM Downscale (9)	≥ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	≤ 108/125 of Full Scale	3(1)
IRM Downscale (3) (8)	≥ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	≤ 1 x 10^5 Counts/Second	1(1) (6)
SRM Detector Not Full In (4) (8)	(≥ 100 cps)	1(1) (6)
SRM Inoperative (8)	(10a)	1(1) (6)
Flow Bias Comparator	≤ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	≤ 110% Recirc. Flow	1
SRM Downscale (8) (7)	≥ 3 Counts/Second	1(1) (6)
RSCS Rod Group C Bypass	≥20% Core Thermal Power	(11)

11. There are two pressure transmitters, which provide the signal to allow Group C rod withdrawal. Failure of either unit will not allow the Group C withdrawal. These units are required to function only until 6500 MWD/T at which time the Group C interlock will be removed
7 after AEC approval.

As described on page 101, the RSCS operation is required below 20% core thermal power. Therefore, the pressure switches will be calibrated to 22% core thermal power to allow for the instrumentation and calculation accuracy.

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TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Flow Bias)	(1) (3)	Once/6 Months	Once/Day
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/operating cycle)	Once/Oper. Cycle (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/Oper. Cycle	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
7 RSCS Rod Group C Bypass	(1) (11)	Once/3 Months	N.A.

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 2.0×10^5; thereafter, according to Figure 4.1.1(after AEC 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, high drywell pressure and high radiation main steam line tunnel 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 1/2 scram) 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. The RSCS Rod Group C Bypass function is required for the first 6500 MWD/T
of the initial core loading. This function is provided by two pressure
7 transducers which sense turbine first stage pressure which is then correlated
with core thermal power. This bypass function assures that control rod
worths are controlled as described in the Bases for Specification 3.3.B.3.

prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety.

7 The RSCS Rod Group C Bypass function is required only during the first 6500 MWD/T of the initial core loading. This function is provided by two pressure transducers which sense turbine first stage pressure which is then correlated with core thermal power. This bypass function assures that control rod worths are controlled as described in the Bases for Specification 3.3.B.3.

The refueling interlocks also operate one logic channel, and are required for a safety only when the mode switch is in the refueling position.

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

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the stack release rate limit given in Specification 2.4.3.a of the Environmental Technical Specifications is not exceeded.

Two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. The trip is actuated by one hi-hi or two downscale indications.

LIMITING CONDITIONS FOR OPERATION

3.7.A (cont'd.)

d. If specifications 3.7.A.4.a,b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours.

5. Oxygen Concentration

a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons.

d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.2 below, both trains of the standby gas treatment system and the diesel generators

SURVEILLANCE REQUIREMENTS

4.7.A (cont'd.)

c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation of the position indication switch. The differential pressure setpoint shall be verified.

d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met. 7

5. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect.

B. Standby Gas Treatment System

1.a. At least once per operating cycle it shall be demonstrated that pressures drop across the combined high efficiency

COOPER NUCLEAR STATION
TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position				Trip Level Setting	Minimum Number of Operable Channels Per Trip System (1)	Action Required When Equipment Operability is Not Assured (1)
	Shutdown	Startup	Refuel	Run			
Mode Switch in shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	≤120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	≤(0.66W+54) $\frac{2.61}{\text{P.F.}}$ (14)	2	A or C
High Flux	X(7)	X (9)	X (9)	(16)	≤15% Rated Power		A or C
Inoperative		X (9)	X (9)	X	(13)	2	A or C
Downscale		(11)		X(12)	≥2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X (9)	X(10)	X	≤1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X (9)	X(8)	X	≤2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	≥ + 12.5 in. indicated level	2	A or D
Scram Discharge Volume High Water Level CRD-LS-231 A,B,C, & D	X (2)(7)	X	X (2)	X	≤36 gallons	2	A

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TABLE 3.1.1 (Page 2)
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position				Trip Level Setting	Minimum Number of Operable Channels Per Trip System (1)	Action Required When Equipment Operability is Not Assured (1)
	Shutdown	Startup	Refuel	Run			
Main Steam Line High Radiation RMP-RM-251, A,B,C, & D		X(9)		X	< 6 Times normal full power back ground	2	A or D
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D		X(3)(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 psi 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
Reactor Pressure Permissive NBI-PS-51 A,B,C, & D		X(9)		X	≤1000 psig	2	A or C
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	≤ 30% first stage press.	2	A or B