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Docket No. 50-298

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 7 to Facility License No. DPR-46 for the Cooper Nuclear Station. This amendment includes Change No. 10 to the Technical Specifications and is in response to those remaining items of your request dated May 28, 1974, not previously reviewed by the Regulatory staff.

This amendment modifies the core spray and primary containment instrumentation requirements and revises the Specifications for filter systems and coolant chemistry.

Copies of the related Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 7
   w/Change No. 10
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

CP

See previous yellow for concurrences

OFFICE	RL:ORB #2	RL:ORB #2	RL:ORB #2	OELD	RL:OR KRG	
SURNAME	RMDiggs	JSapir/tc	DLZiemann		KRGoller	
DATE	11/ /74	11/ /74	11/ /74	12/ /74	2/6 /74	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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Columbus, Nebraska 68601

Gentlemen:

The Commission has issued the enclosed Amendment No. 7 to Facility License No. DPR-46 for the Cooper Nuclear Station. This amendment includes Change No. 10 to the Technical Specifications and is in response to those remaining items of your request dated May 28, 1974, not previously reviewed by the Regulatory staff.

? || The amendment replaces several sections of the Technical Specifications with more recent formulations and updates instrumentation requirements.

Copies of the related Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 7
   w/Change No. 10
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OFFICE	L:ORB #2	L:ORB #2	L:ORB # 2	OGC R. Kinsey	L:OR	
SURNAME	RMDiggs/tlc	JSapir	DLZiemann	IA STRIDIRON	KRGoller	
DATE	10/7/74	10/7/74	10/11/74	12/10/74	10/ /74	

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12/10/74

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Nemaha County Board of Commissioners
Nebraska County Courtroom
Auburn, Nebraska 68305

Mr. James L. Higgins, Director
Department of Environmental Control
Executive Building, 2nd Floor
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Environmental Protection Agency
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Kansas City, Missouri 64108

Auburn Public Library
1118 - 15th Street
Auburn, Nebraska 68305

OFFICE➤						
SURNAME➤						
DATE➤						

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
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 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:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

"(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. 13."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

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

Attachment:
Change No. 10 to the
Technical Specifications

Date of Issuance: FEB 06 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 10 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 of Facility License No. DPR-46 are hereby changed by replacing pages 29, 53, 55, 61, 62a, 63, 65, 70, 80, 134, 135, 147, 148, 164, 165, 181, 182, 183, 210, 211, 213 and 214 with the attached revised pages bearing the same numbers and with additional pages 148a, 165a, 183a, 210a and 214a. Changed areas on these pages are shown by marginal lines.

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 (
	Shutdown	Startup	Refuel	Run			
Main Steam Line High Radiation RMP-RM-251, A,B,C, & D		X(9)		X	< 3 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

COOPER NUCLEAR STATION
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
Core Spray Pump Time Delay	CS-TDR-K16 A & B	$9 \leq T \leq 11$ seconds	1	B
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

10

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	$\geq$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
10	(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	$\leq$.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 27 X 2/1F	Loss of Voltage Loss of Voltage	1 1	B B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G 27 X 2/1G	Loss of Voltage Loss of Voltage	1	B
Pump Discharge Line Low Pressure	CM-PS-266	$\geq$5 psig	(3)	D

COOPER NUCLEAR STATION

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)	$\leq$(0.66W + 42) (2)	2(1)
APRM Upscale (Startup)	$\leq$12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$(0.66W + 41) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3) (8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1) (6)
SRM Detector Not Full In (4) (8)	($\geq$ 100 cps)	1(1) (6)
SRM Inoperative (8)	(10a)	1(1) (6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8) (7)	$\geq$ 3 Counts/Second (0.3 Counts/Second prior to achieving burnup of 3500 MWD/T on first core)	1(1) (6)
RSCS Rod Group C Bypass	$\geq$ 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 after AEC approval.

10

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.

COOPER NUCLEAR STATION

TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I.D. No.	Setting Limit	Number of Sensor Channels Provided by Design	Action (1)
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	(2)	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	<100 mr/hr	2	B
Liquid Radwaste Discharge Isolation	RMV-RM-2	(3)	1	C
Main Control Room Ventilation Isolation	(RMV-RM-1)	4×10^3 CPM	1	D
Mechanical Vacuum Pump Isolation	RMP-RM-251 A-D	3 times normal full power background. Alarm at 1.5 times normal full power background.	4	E

10

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.

   A. Refer to Section 2.4.3.a.7 of the Environmental Technical Specifications.

   B. Cease refueling operations, isolate secondary containment and start SBGT.

   C. Refer to Sections 2.4.1.b. of the Environmental Technical Specifications.

   D. Refer to Section 3.11.A.

   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 setting to correspond to Specification 2.4.3.a.1 of the Environmental Technical Specifications.

3. Trip setting to correspond to Specification 2.4.1.b.1 of the Environmental Technical Specifications.

COOPER NUCLEAR STATION
TABLE 3.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION

Instrument	Instrument I:D. No.	Range	Minimum Number of Operable Instrument Channels	Action Required Whe Minimum Condition I Satisfied (1)
Reactor Water Level	NBI-LI-85A NBI-LI-85B	-159" to +60" -150" to +60"	2	A,B,C
Reactor Pressure	RFC-PI-90A RFC-PI-90B	0 - 1200 psig 0 - 1200 psig	2	A,B,C
Drywell Pressure	PC-PI-512A PC-PR-512B	0 - 80 psia 0 - 80 psia	2	A,B,C
Drywell Temperature	PC-TR-503 PC-TI-505	50 - 170°F 50 - 350°F	2	A,B,C
Suppression Chamber Air Temperature	PC-TR-21A PC-TA-20A,C	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Temperature	PC-TR-21B PC-TA-20B,D	0 - 300°F 0 - 400°F	2	A,B,C 10
Suppression Chamber Water Level	PC-LI-10 PC-LR-11 PC-LI-12	(-4' to +6') (-4' to +6') -10" to +10"	2	A,B,C
Suppression Chamber Pressure	PC-PR-20 PC-PR-512B	0 - 2 psig 0 - 80 psia	2	B,C A,B,C
Control Rod Position	N.A.	Indicating Lights	1	A,B,C,D
Neutron Monitoring	N.A.	S.R.M., I.R.M., LPRM 0 - 100% power	1	A,B,C,D

COOPER NUCLEAR STATION

TABLE 4.2.B (Page 1)

CORE SPRAY SYSTEM TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument				
1. Reactor Low Water Level	NBI-LIS-72, A,B,C & D	Once/Month (1)	Once/3 Months	Once/Day
2. Reactor Low Pressure	NBI-PS-52, A & C NBI-PIS-52, B & D	Once/Month (1)	Once/3 Months	None
3. Drywell High Pressure	PC-PS-101, A,B,C & D	Once/Month (1)	Once/3 Months	None
4. Core Spray Pump Disch. Press.	CS-PS-44, A & B CS-PS-37, A & B	Once/Month (1) Once/Month (1)	Once/3 Months Once/3 Months	None None
5. Core Spray Pump Time Delay	CS-TDR - K16, A & B	Once/Month (1)	Once/Oper. Cycle (4)	None
10				
6. Emergency Bus Low Volt Relay	27X1 - 1F & 1G 27X2 - 1F & 1G	Once/Oper. Cycle Once/Oper. Cycle	Once/5 Years Once/5 Years	None None
7. Aux. Bus Low Voltage Relay	27X3 - 1A & 1B	Once/Oper. Cycle	Once/5 Years	None
8. Pump Disch. Line Low Press.	CM-PS-73, A & B	Once/3 Months	Once/3 Months	None
Logic (4) (6)				
1. Logic Power Monitor		Once/6 Months	N.A.	N.A.
2. Core Spray Initiation		Once/6 Months	N.A.	N.A.
3. Pump & Valve (Signal Override) Control		Once/6 Months	N.A.	N.A.

COOPER NUCLEAR STATION
TABLE 4.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION
TEST AND CALIBRATION FREQUENCIES

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A	Once/6 Months	Each Shift
	NBI-LI-85B	Once/6 Months	Each Shift
Reactor Pressure	RFC-PI-90A	Once/6 Months	Each Shift
	RFC-PI-90B	Once/6 Months	Each Shift
Drywell Pressure	PC-PI-512A	Once/6 Months	Each Shift
	PC-PR-512B	Once/6 Months	
Drywell Temperature	PC-TR-503	Once/6 Months	Each Shift
	PC-TI-505	Once/6 Months	Each Shift
Suppression Chamber Air Temperature	PC-TR-21A	Once/6 Months	Each Shift
	PC-TA-20B,D	Once/6 Months	Each Shift
Suppression Chamber Water Temperature	PC-TR-21B	Once/6 Months	Each Shift
	PC-TIS-20B,D	Once/6 Months	Each Shift 10
Suppression Chamber Water Level	PC-LI-10	Once/6 Months	Each Shift
	PC-LR-11	Once/6 Months	Each Shift
	PC-LI-12	Once/6 Months	Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift
	PC-PR-512B	Once/6 Months	Each Shift
Control Rod Position	N.A.	N.A.	Each Shift
Neutron Monitoring (APRM)	N.A.	Once/Week	Each Shift

3.6.B (cont'd)

2. Prior to startup and during the operation of the reactor up to 10% of rated power, and during hot standby, the reactor coolant shall not exceed the following limits.

a. Conductivity 2 μmho/cm at 25°C

b. Chloride 0.1 ppm

3. During reactor operation in excess of 10% of rated power, the reactor coolant shall not exceed the following limits:

a. Conductivity 1 μmho/cm at 25°C

b. Chloride 0.2 ppm

4. During the reactor operation in excess of 10% of rated power, the reactor coolant may exceed
10 the limits of Paragraph 3.6.B.3 only for the time limits specified here. If these time limits or the following maximum limits are exceeded, the reactor shall be shutdown immediately and placed in the cold shutdown condition.

a. Conductivity Time above 1 μmho/cm at 25°C, 2 weeks/year Maximum limit-10 μmho/cm at 25°C

b. Chloride Time above 0.2 ppm, 2 weeks/year Maximum limit-0.5 ppm

The reactor shall be shut down if pH is <5.6 or >8.6 for a 24-hour period.

5. When the reactor is not pressurized (i.e. at or below 212°F), reactor coolant shall be maintained below the following limits:

a. Conductivity 10 μmho/cm at 25°C

b. Chloride 0.5 ppm

4.6 (cont'd)

b. If the gross activity counts of a sample indicate an activity concentration above 3.1 uCi/gm of dose equivalent I-131, an isotopic analysis shall be performed and quantitative measurements made to determine the dose equivalent I-131 concentration.

c. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. Reactor coolant shall be continuously monitored for conductivity.

3. Prior to startup, during the operation of the reactor and during hot standby, a sample of the reactor coolant shall be analyzed.

a. At least every 80 hours for conductivity and chloride ion content
10 when the continuous conductivity monitor reading is $\leq$0.7 μmho/cm 25°C.

b. At least every 24 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >0.7 $\leq$2.0 μmho/cm at 25°C.

c. At least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >2 but $\leq$3.5 μmho/cm at 25°C.

d. At least every 4 hours for conductivity, chloride ion content, and pH, when the continuous conductivity monitor reading is >3.5 μmho/cm at 25°C or when the continuous conductivity monitor is inoperable.

4. When the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 80 hours for conductivity and chloride ion content.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. <u>Coolant Leakage</u>	C. <u>Coolant Leakage</u>
1. Any time irradiated fuel is in the the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the identified reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.
2. The sump flow measuring systems, containment atmospheric radiation monitor and the containment atmospheric sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 30 days unless the system is made operable sooner.	
3. If the specifications 3.6.C.1 & 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	

As described in the safety analysis report, detailed stress analyses have been made on the reactor vessel for both steady-state and transient conditions with respect to material fatigue. The results of these analyses are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145° F.

The maximum calculated neutron fluence of 1 Mev or greater, based on 100 percent rated power and 100 percent availability for 40 years, is 8.5×10^{17} nvt. The neutron flux wires are removed and tested after approximately one year of operation during the first refueling outage to experimentally verify the calculated values of integrated neutron flux. The RT NDT is determined by utilizing the value of the fluence measured at the core mid-plane level. This approach is conservative because the fluence level decreases as the point of measurement is removed from the core mid-plane level. In addition, vessel material samples will be located within the vessel to monitor the effect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal and heat affected zone metal. These samples will receive neutron exposure more rapidly than the vessel wall material and therefore, will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the Shift in /RT NDT used in the specification is conservative.

B. Coolant Chemistry

Materials in the primary system are primarily Type-304 stainless steel and Zircaloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel. 10

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams[1], where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking. These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and therefore, are of limited significance to BWR conditions. They are, however, a qualitative indication of trends.

[1] W. L. Williams, Corrosion 13, 1957, p. 539t.

The water quality specification is further supported by General Electric stress Corrosion test data summarized as follows:

1. Type-304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at pH7. Test specimens were bent beam strips stressed over their yield strength. After 2100 hours exposure, no cracking or failures occurred.

2. Welded Type-304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at pH7. Uniaxial tensile test specimens were stressed at 125% of their 550°F yield strength. No cracking or failure occurred at 15,000 hours exposure.

Zircaloy and Inconel alloys do not exhibit chloride stress corrosion cracking failure mechanisms.

When conductivity is in its normal range, pH, chloride and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt which
10 would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWR's, however, where no additives are used and where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods are available to the operator for correcting the off-standard condition including operation of the reactor cleanup system, reducing the input of impurities, and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to re-establish the purity of the reactor coolant.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken periodically serve as a reference for calibration of these monitors and are considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges.

During reactor startup and hot standby, the dissolved oxygen content of reactor water may be higher than during normal power operation. During this period more restrictive limits are established. After power operation has been established, boiling deaerates the reactor water reducing the influence of oxygen on potential chloride stress corrosion cracking.

The relationship of chloride concentration to specific conductance measures at 25°C for chloride compounds such as sodium chloride and hydrochloric acid can be calculated.[2] Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships. The sampling frequency when reactor water has a low specific conductance is adequate for calibration and routine audit purposes. When specific conductance increases, and higher
10 chloride concentrations are possible, or when continuous conductivity monitoring is unavailable, increased sampling is provided.

Isotopic analysis to determine major contributors to activity will be performed by a gamma scan. The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 30 rem at the exclusion distance during the 2-hour period following a steam line break. This dose is computed with the assumptions of a closure of the isolation valves within 5 seconds (closure time as required on Table 3.7.1 of these specifications) and a X/Q value of 3.0×10^{-4} Sec/M.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

C. Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study

10 [2] J.M. Skarpelos and J. W. Bagg, Chloride Control in BWR Coolants, June 1973, NEDO-10899

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

10 1. Except as specified in 3.7.B.3 below, both circuits 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.

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. At least once per operating cycle the following conditions shall be demonstrated. 10

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is

3.7.B (cont'd)

required for operation of such circuits shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon

b. The results of laboratory carbon sample analysis shall show ≥95% radioactive methyl iodide removal at a velocity within 20 percent of actual system design, 0.5 to 1.5 mg/m^3 inlet methyl iodide concen-
10 tration, ≥70% R.H. and ≥190°F.

c. Fans shall be shown to operate within ± 10% design flow.

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

4. If these conditions cannot be met, procedures shall be initiated immediately to establish reactor conditions for which the standby gas treatment system is not required.

4.7.B (cont'd)

less than 6 inches of water at the system design flow rate.

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

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system. 10

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

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

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

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

4.a. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

C. Secondary Containment

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

4.7.B (cont'd)

b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one circuit of the standby gas treatment system becomes inoperable
10 the other circuit shall be demonstrated to be operable immediately and daily thereafter.

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

Drywell Interior

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Drywell to Suppression Chamber Structure Leakage Test

A leakage test of the drywell to suppression chamber structure shall be conducted at the end of each refueling outage. Using the drywell purge and vent system, the drywell pressure will be increased by 1.0 psi with respect to the wetwell pressure and held constant. Maintaining 1 psig in the drywell, the wetwell pressure will be monitored with a sensitive mercury manometer.

The maximum allowable bypass leakage limit is equivalent to a rate of change of wetwell pressure less than 0.30 inches of water per minute as measured over a ten minute period with a differential pressure of 1 psid between the drywell and the wetwell.

In the event the rate of change of pressure exceeds this value, the source of leakage will be identified and eliminated before power operation is resumed.

3.7.B&C. Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation. When the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust the reactor
building atmosphere to the stack during secondary containment isolation
conditions. Both standby gas treatment system fans are designed to
automatically start upon containment isolation and to maintain the reactor 10
building pressure to the design negative pressure so that all leakage
should be in-leakage. Should one system fail to start, the redundant
system is designed to start automatically. Each of the two fans has 100
percent capacity.

3.7.B & 3.7.C BASES (c 'd)

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of COP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of 10
at least 95 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

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

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

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

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability. 10

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

4.7.B & 4.7.C BASES

with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52.

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

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

If system drains are present in the filter/adsorber banks, loop-seals 10
must be used with adequate water level to prevent by-pass leakage from the banks.

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

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

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

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

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

LIMITING CONDITIONS FOR OPERATION

3.11 Additional Safety Related Plant Capabilities

Applicability:

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

Objective:

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

A. Main Control Room Ventilation

1. Except as specified in Specification 3.11.A.3 below, the control room air treatment system, the diesel generators required for operation of this system and the main control room
10 air radiation monitor shall be operable at all times when containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq 99\%$ DOP removal and $\geq 99\%$ halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq 90\%$ radioactive methyl iodide removal at a velocity within 20% of system design, 0.05 to 0.15 mg/m^3 inlet iodide concentration, $\geq 95\%$ R.H. and $\geq 125°F$.

c. Fans shall be shown to operate within $\pm$ 10% design flow.

SURVEILLANCE REQUIREMENTS

4.11 Additional Safety Related Plant Capabilities

Applicability:

Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water system and the service water system which are required by the corresponding Limiting Conditions for Operation.

Objective:

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

A. Main Control Room Ventilation

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow
10 rate.

2.a. The tests and sample analysis of Specification 3.11.A.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each circuit shall be operated at least 10 hours every month.

LIMITING CONDITIONS FOR OPERATION

3. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor or refueling operations are permissible only during the succeeding seven days unless such circuit is sooner made operable.

4. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours. If
10 refueling operations are in progress, such operations shall be terminated in an orderly manner.

SURVEILLANCE REQUIREMENTS

3. At least once per operating cycle 10
automatic initiation of the system shall be demonstrated.

LIMITING CONDITIONS FOR OPERATION

B. Reactor Building Closed Cooling Water System (REC)

1. Both reactor building closed cooling water loops and their associated pumps shall be operable whenever irradiated fuel is in the vessel or the spent fuel pool, except as specified in 3.11.B.2 and 3.11.B.3 below.

2. From and after the date that any component in one loop becomes inoperable continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all the components of the other loop and the active components of the engineered safeguards compartment cooling systems, the diesel generator associated with the operable loop are operable.

   The allowable repair time does not apply when the reactor is in the shutdown mode and reactor pressure is less than 75 psig.

3. Both reactor building closed cooling water loops with one pump per loop shall be operable as stated in 3.11.B.1 and 3.11.B.2 above during reactor head-off operations requiring LPCI or Core Spray System availability or service water cooling shall be available.

4. If the requirements of 3.11.B.1 through 3.11.B.3 cannot be met, the reactor shall be shutdown in an orderly manner and in the Cold Shutdown condition within 24 hours or operations requiring LPCI or core spray system availability shall be halted.

SURVEILLANCE R[illegible]IREMENTS

B. Reactor Building Closed Cooling Water System (REC)

1. REC System Testing

	Item	Frequency
a.	Pump Operability	Once/Month
b.	Motor operated Valve Operability	Once/Month
c.	Pump flow rate Each pump shall deliver 1175 gpm at 65 psid.	Once/3 months and after pump maintenance
d.	System head tank level shall be monitored	Daily

2. When it is determined that any active component in an REC loop is inoperable, all components in the other loop shall be demonstrated operable immediately and weekly thereafter.

3.11 BASES

A. Main Control Room Ventilation System

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

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal 10
adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is not immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours, or refueling operations are terminated.

B. Reactor Building Closed Cooling Water System

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

The system has additional flexibility provided by the capability of interconnection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of LPCI or the core spray systems.

C. Service Water System

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

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

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

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

D. Battery Room Ventilation

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

4.11 BASES

A. Main Control Room Ventilation System

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

Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant should be performed in accordance with USAEC Report DP-1082. 10

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52.

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

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

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. DPR-46

(CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS)

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

INTRODUCTION

In a letter dated May 28, 1974, the Nebraska Public Power District (NPPD) transmitted a series of proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). In response to staff questions dated August 1, 1974, NPPD submitted additional information in a letter dated September 16, 1974. Portions of the NPPD request have been addressed in Commission actions dated August 15, 1974, October 9, 1974, and October 15, 1974. This safety evaluation discusses the remaining items not previously reviewed.

DISCUSSION AND EVALUATION

The eleventh proposed change would reduce the operability requirement on the actuation instrumentation for the core spray minimum flow valve. Whereas the current specifications allow 24 hours to repair a defective channel before declaring the core spray system inoperable, the proposed specification would merely require repair as soon as possible. A low flow bypass line is provided from the core spray pump discharge to prevent the pump from overheating when pumping against a closed discharge valve. When the flow reaches a value large enough to assure safe pump operation, the minimum flow bypass valve is automatically closed directing all flow into the reactor vessel. NPPD has reported that onsite testing has shown that each core spray pump will deliver the required system flow with the bypass valves fully open. In response to the staff's question, NPPD has verified that the position of the bypass valve does not affect these test conclusions. Since the design requirements of the core spray subsystem are met with the bypass valve fully open, the staff has concluded that failure of the valve to close does not diminish the capability of the core spray system to perform its safety function nor reduce the safety

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DATE➤						

4.11 BASES (cont'd)

Demonstration of the automatic initiation capability is necessary to assure system performance capability. 10

B. Reactor Building Closed Cooling Water System

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

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

C. Service Water System

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

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

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

D. Battery Room Ventilation

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-46 issued to the 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 modifies the core spray and primary containment instrumentation requirements and revises the Specifications for filter systems and coolant chemistry.

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

For further details with respect to this action, see (1) the application for amendment dated May 28, 1974, and supplement thereto dated September 16, 1974, (2) Amendment No. 7 to License No. DPR-46, with Change No. 10 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 Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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

margins presently incorporated in the Technical Specifications and operability of the valve actuation instrumentation is not required. Therefore, the proposed change has been modified to remove the requirement for this channel. This change is implemented by revisions to the Tables on pages 53 and 70.

The fifteenth and sixteenth proposed changes would incorporate clarifications for the standby gas treatment system and main control room ventilation system specifications. The staff has recently developed improved Technical Specifications and Bases for these systems. These new specifications more clearly define the limiting conditions for operation and surveillance requirements and incorporate requirements to assure that under accident conditions the filters in these systems will meet or exceed the efficiencies assumed in the design basis accident analysis. The associated bases reflect acceptable methods and standards for surveillance of the filter and for replacement of the filter adsorber media. NPPD has been informed of these specifications and has agreed to their incorporation. The staff has concluded that the new specifications and bases increase plant safety and has, therefore, incorporated them in Sections 3.7.B and 3.11.A of the Cooper Nuclear Station Technical Specifications.

The seventeenth proposed change would replace specifications on coolant chemistry control with those proposed in the General Electric Standard Safety Analysis Report (GESSAR), Docket 50-447. We have reviewed these specifications and have concluded that the proposed maximum contaminant levels as well as the proposed pH and conductivity requirements have been shown by tests and service experience to adequately protect against corrosion and stress corrosion problems, and are in accordance with requirements of Regulatory Guide 1.56. We have corrected several typographical errors in the proposed request and made an editorial change which more clearly and explicitly states the action to be taken in case a limiting condition for operation is exceeded. These modifications do not affect plant safety. The staff has 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 replacing pages 134, 135, 147 and 148 with revised pages 134, 135, 147, 148 and 148a.

The nineteenth proposed change would modify the specifications describing the primary containment surveillance instrumentation. As a result of discussions with the NPPD staff, we have modified the proposed change to eliminate several discrepancies. The change, as modified, identifies the specific instrument channels and ranges monitoring the suppression chamber air temperature, reduces the range on an existing suppression chamber water temperature monitor from 400°F to 300°F and adds new suppression chamber water temperature and water level channels. In response to a staff question, NPPD stated, and we agree, that the proposed instrument ranges are adequate under all operating, transient and accident conditions. The additional channels will provide increased assurance that the primary containment parameters are properly monitored. The staff concludes that the proposed changes do not reduce the safety provisions of the Technical Specifications and find them acceptable. The changes are implemented by revisions on pages 65 and 80.

Five pages have been reissued to correct previous administrative errors. Changes No. 4 (April 18, 1974) and No. 7 (October 15, 1974) inadvertently omitted several changes previously incorporated into the Technical Specifications. These errors have been corrected with the reissuance of revised pages 29, 61 and 63. Revised page 55 has been reissued to correct an omission of a change addressed and accepted in the safety evaluation accompanying Change No. 7. Revised page 62a has been reissued to replace a reference to a page number with a reference to the desired specification number.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) 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, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: FEB 06 1975

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

OK for final

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. DPR-46

(CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS)

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

INTRODUCTION

In a letter dated May 28, 1974, the Nebraska Public Power District (NPPD) transmitted a series of proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). In response to staff questions dated August 1, 1974, NPPD submitted additional information in a letter dated September 16, 1974. Portions of the NPPD request have been addressed in Commission actions dated August 15, 1974, October 9, 1974, and October 15, 1974. This safety evaluation discusses the remaining items not previously reviewed.

DISCUSSION AND EVALUATION

actuation instrumentation for the valve.

The eleventh proposed change would reduce the operability requirement on ~~for~~ the core spray minimum flow instrumentation. Whereas the current specifications allow 24 hours to repair a defective channel ~~circuit~~ before declaring the core spray system inoperable, the proposed specification would merely require repair as soon as possible. A low flow bypass line is provided from the core spray pump discharge to prevent the pump from overheating when pumping against a closed discharge valve. When the flow reaches a value large enough to assure safe pump operation, the minimum flow bypass valve is automatically closed directing all flow into the reactor vessel. NPPD has reported that onsite testing has shown that each ~~both~~ core spray pump~~s~~ will deliver the required system flow with the bypass valves fully open. In response to the staff's question, NPPD has verified that the position of the bypass valve does not affect these flow measurements. Since the design requirements of the core spray subsystem are met with the bypass valve fully open, the staff has concluded that failure of the valve to close does not diminish the capability of the core spray system to perform its safety function nor reduce the safety margins presently on a high priority basis.

The valve

and operability of [illegible] instrumentation is not required

incorporated in the Technical Specifications. The proposed change is therefore ~~found~~ acceptable and implemented by appropriate revisions to Table 3.2.B on page 53 of the Technical Specifications.

The fifteenth and sixteenth proposed changes would incorporate clarifications for the standby gas treatment system and main control room ventilation system specifications. The staff has recently developed improved Technical Specifications and Bases for these systems. These new specifications more clearly define the limiting conditions for operation and surveillance requirements and incorporate requirements to assure that under accident conditions the filters in these systems will meet or exceed the efficiencies assumed in the design basis accident analysis. The associated bases reflect acceptable methods and standards for surveillance of the filter and for replacement of the filter absorber media. The staff has concluded that the new specifications and bases increase plant safety and has, therefore, incorporated them in Sections 3.7.B and 3.11.A of the Cooper Nuclear Station Technical Specifications.

cleared with Licensee [illegible]

NPPD has been informed of these specifications and has agreed to their incorporation.

The seventeenth proposed change would replace specifications on coolant chemistry control with those proposed in the General Electric Standard Safety Analysis Report (GESSAR), Docket 50-447. We have reviewed these specifications and have concluded that the proposed maximum contaminant levels as well as the proposed pH and conductivity requirements have been shown by tests and service experience to adequately protect against corrosion and stress corrosion problems, and are in accordance with requirements of Regulatory Guide 1.56. ~~We have made an editorial change in the proposed request.~~ The staff has 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 replacing pages 134, 135, 147 and 148 with revised pages 134, 135, 147, 148 and 148a.

? editorial?

Insert C

The nineteenth proposed change would modify the primary containment surveillance instrumentation. ~~to correspond with existing instruments.~~ ~~Several channels were added and the range on one instrument was reduced.~~ a suppression chamber water temperature monitor was changed from 400°F to 300°F. In response to a staff question, NPPD stated, and we agree, that the proposed instrument ranges are adequate under all operating, transient and accident conditions. The additional channels will provide increased assurance that the primary containment parameters are properly monitored. The staff concludes that the proposed changes do not reduce the safety provisions of the Technical Specifications and find them acceptable. The changes are implemented by revisions on pages 65 and 80.

B ?

To correct minor errors on ~~four~~ five previously issued pages in Change No. 4 (April 18, 1974) and Change No. 7 (October 15, 1974), also contained as part of this action are corrected pages 29, 55, 61, 62a and 63.

typo?

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) 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, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: