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December 17, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4

Re: Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 50-414
Proof and Review Technical Specifications

Dear Mr. Denton:

In response to your October 10, 1985 letter which transmitted the Proof and Review Technical Specifications for Catawba Units 1 and 2 and as a supplement to my letters of October 30, 1985 and November 7, 1985, attached are additional corrections to errors found by our ongoing review.

If you have any questions regarding this response please contact Mr. Roger W. Ouellette at (704)373-7530.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

RWO:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Resident Inspector
Catawba Nuclear Station

ADD: PWR - A/BC's TECH SUPPORT

AD - J. KNIGHT (ltr only)
EB (BALLARD)
EICSB (ROSA)
PSB (GAMMILL)
RSB (BERLINGER)
FOB (BENAROYA)

B021
1/1

20 Mech. Pene. Room JJ-62 El. 543 + 0 3 0 3 A

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE*			FUNCTION**
				SMOKE	FLAME	HEAT	
1	R.H.R. Pump 1B	GG-53	El.522 + 0	1	0	1	A
2	R.H.R. Pump 1A	FF-53	El.522 + 0	1	0	1	A
3	Cont. Spray Pump 1B	GG-54	El.522 + 0	3	0	3	A
4	Cont. Spray Pump 1A	GG-55	El.522 + 0	2	0	2	A
5	R.H.R. Pump 2B	GG-61	El.522 + 0	1	0	1	A
6	R.H.R. Pump 2A	FF-61	El.522 + 0	1	0	1	A
7	Cont. Spray Pump 2B	GG-60	El.522 + 0	3	0	3	A
8	Cont. Spray Pump 2A	GG-59	El.522 + 0	2	0	2	A
9	Aux. F. W. Pumps	BB-51	El.543 + 0	14	0	12(6)	A(B)
10	Mech. Pene. Room	JJ-52	El.543 + 0	3	0	3	A
11	Corridor/Cables	NN-51	El.543 + 0	6	0	6	A
12	Recip. Chg. Pump	JJ-53	El.543 + 0	1	0	1	A
13	Safety Inj Pump 1B	HH-53	El.543 + 0	1	0	1	A
14	Safety Inj Pump 1A	GG-53	El.543 + 0	1	0	1	A
15	Cent. Chg. Pump 1B	JJ-54	El.543 + 0	2	0	2	A
16	Cent. Chg. Pump 1A	JJ-55	El.543 + 0	2	0	2	A
17	Aisles/Cables	KK-56	El.543 + 0	18	0	18	A
18	Aisles/Cables	EE-55	El.543 + 0	6	0	6	A
19	AFW Pumps (Unit 2)	BB-63	El.543 + 0	14	0	12(6)	A(B)
21	Aisles/Cables	NN-63	El.543 + 0	6	0	6	A
22	Recip. Chg. Pump	JJ-60	El.543 + 0	1	0	1	A
23	Safety Inj. Pump 2B	HH-60	El.543 + 0	1	0	1	A
24	Safety Inj. Pump 2A	GG-60	El.543 + 0	1	0	1	A
25	Cent. Chg. Pump 2B	JJ-59	El.543 + 0	2	0	2	A
26	Cent. Chg. Pump 2A	JJ-58	El.543 + 0	2	0	2	A
27	Aisles/Cables	KK-59	El.543 + 0	20	0	20	A
28	Aisles/Cables	EE-58	El.543 + 0	6	0	6	A
29	SW Gear Equip. Room	AA-50	El.560 + 0	7	0	0	A
30	Elect. Pene. Room	CC-50	El.560 + 0	8	0	0	A
31	Corridor/Cables	EE-53	El.560 + 0	5	0	5	A
32	Corridor/Cables	KK-52	El.560 + 0	8	0	8	A
33	Corridor/Cables	NN-54	El.560 + 0	10	0	10	A

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{44 Corridors/Cables KK-63 El. 560 + 0 8 0 8 A }

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
34	Aisles/Cables	JJ-56	El.560 + 0	14	0	14	A
35	Motor Control Centers	GG-56	El.560 + 0	2	0	2	A
36	Cable Tray Access	FF-56	El.568 + 0	2	0	2	A
37	Equip. Batteries	DD-55	El.554 + 0	5	0	2	A
38	Equip. Batteries	CC-55	El.554 + 0	5	0	4	A
39	Battery Room	CC-56	El.554 + 0	5	0	4	A
41	SW Gear Equip. Room	AA-64	El.560 + 0	17	0	0	A
42	Elect. Pene. Room	CC-65	El.560 + 0	7	0	0	A
43	Corridor/Cables	FF-61	El.560 + 0	8	0	0	A
45	Aisles/Cables	NN-60	El.560 + 0	5	0	5	A
46	Aisles/Cables	HH-59	El.560 + 0	13	0	13	A
47	Motor Control Center	GG-58	El.560 + 0	13	0	13	A
48	Cable Tray Access	FF-58	El.560 + 0	2	0	2	A
49	Equip. Batteries	DD-60	El.560 + 0	2	0	2	A
50	Equip. Batteries	CC-60	El.560 + 0	5	0	4	A
51	Battery Room	CC-59	El.560 + 0	5	0	4	A
53	SW Gear Equip. Room	AA-49	El.577 + 0	17	0	0	A
54	Aisles/Cables	CC-50	El.577 + 0	7	0	0	A
55	Aisles/Cables	NN-52	El.577 + 0	10	0	0	A
56	Aisles/Cables	PP-55	El.577 + 0	9	0	9	A
57	Aisles/Cables	LL-55	El.577 + 0	13	0	13	A
58	Aisles/Cables	HH-55	El.577 + 0	11	0	11	A
59	Motor Control Center	EE-54	El.577 + 0	21	0	21	A
60	Cable Room	CC-56	El.574 + 0	2	0	2	A
62	SW Gear Equip. Room	AA-64	El.577 + 0	18	0	15	A
63	Elect. Pene. Room	CC-64	El.577 + 0	7	0	0	A
64	Aisles/Cables	PP-62	El.577 + 0	10	0	0	A
65	Aisles/Cables	PP-59	El.577 + 0	9	0	9	A
66	Aisles/Cables	LL-59	El.577 + 0	16	0	16	A
67	Aisles/Cables	HH-59	El.577 + 0	11	0	11	A
68	Motor Control Center	FF-60	El.577 + 0	21	0	21	A
69	Cable Room	CC-59	El.577 + 0	2	0	2	A
				18	0	15	A

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION	MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
71	Elect Pene. Room	CC-51 El.594 + 0	10	0	0	A
72	Control Room	CC-56 El.594 + 0	~~23~~ 25	0	6	A
73	Vent. Equip. Room	FF-56 El.594 + 0	9	0	0	A
74	Aisles/Cables	LL-56 El.594 + 0	25	0	25	A
76	Aisles/Cables	PP-54 El.594 + 0	15	0	15	A
79	Elect. Pene. Room	BB-63 El.594 + 0	11	0	0	A
80	Control Room	BB-59 El.594 + 0	22	0	6	A
81	Ven. Equip. Room	FF-58 El.594 + 0	12	0	0	A
82	Aisles/Cables	KK-58 El.594 + 0	28	0	27	A
84	Aisles/Cables	NN-58 El.594 + 0	17	0	17	A
89	Fuel Pool Area #1	PP-50 El.605 + 10	19	7	19	A
90	Fuel Pool Area (Unit 2)	PP-64 El.605 + 10	19	7	19	A
128	UHI Bldg.	HH-44 El.550 + 0	2	3	2	A
129	Fuel Pool Purge Room	NN-50 El.631 + 6	6	0	6	A
130	UHI Bldg. (Unit 2)	HH-71 El.594 + 0	2	3	2	A
131	Reactor Bldg.	0°-45° Bel. El.565 + 3	4	0	0	A
132	Reactor Bldg.	45°-90° Bel. El.565 + 3	3	0	0	A
133	Reactor Bldg.	90°-135° Bel. El.565 + 3	4	0	0	A
134	Reactor Bldg.	135°-180 Bel. El.565 + 3	5	0	0	A
135	Reactor Bldg.	180°-225° Bel. El.565 + 3	4	0	0	A
136	Reactor Bldg.	270°-315° Bel. El.565 + 3	3	0	0	A
137	Reactor Bldg.	315°-0° Bel. El.565 + 3	8	0	0	A
138	Reactor Bldg.	0°-45° Bel. El.586 + 3	6	0	0	A
139	Reactor Bldg.	45°-90° Bel. El.586 + 3	4	0	0	A
140	Reactor Bldg.	90°-135° Bel. El.565 + 3	3	0	0	A
141	Reactor Bldg.	135°-180 Bel. El.586 + 3	8	0	0	A
142	Reactor Bldg.	180°-225° Bel. El.586 + 3	5	0	0	A
143	Reactor Bldg.	315°-0° Bel. El.586 + 3	5	0	0	A
144	Reactor Bldg.	0°-45° Bel. El.593 + 2½	14	0	0	A
145	Reactor Bldg.	45°-90° Bel. El.593 + 2½	17	0	0	A
146	Reactor Bldg.	90-135° Bel. El.593 + 2½	11	0	0	A
147	Reactor Bldg.	135°-180° Bel. El.593 + 2½	10	0	0	A

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE*			
				SMOKE	FLAME	HEAT	FUNCTION**
148	Reactor Bldg.	180°-225°	Bel. El.593 + 2½	2	0	0	A
149	Reactor Bldg.	315°-0°	Bel. El.593 + 2½	7	0	0	A
150	Reactor Bldg (Unit 2)	0°-45°	Bel. El.565 + 3	4	0	0	A
151	Reactor Bldg (Unit 2)	45°-90°	Bel. El.565 + 3	3	~~1~~ 0	~~18~~ 0	A
152	Reactor Bldg (Unit 2)	90°-135°	Bel. El.565 + 3	4	~~2~~ 0	~~2~~ 0	A
153	Reactor Bldg (Unit 2)	135°-180°	Bel. El.565 + 3	5	0	0	A
154	Reactor Bldg (Unit 2)	180°-225°	Bel. El.565 + 3	3	0	0	A
155	Reactor Bldg (Unit 2)	270°-315°	Bel. El.565 + 3	4	0	0	A
156	Reactor Bldg (Unit 2)	315°-0°	Bel. El.565 + 3	6	0	0	A
157	Reactor Bldg (Unit 2)	0°-45°	Bel. El.586 + 6	6	0	0	A
158	Reactor Bldg (Unit 2)	45°-90°	Bel. El.586 + 6	4	0	0	A
159	Reactor Bldg (Unit 2)	90°-135°	Bel. El.586 + 6	3	0	0	A
160	Reactor Bldg (Unit 2)	135°-180°	Bel. El.586 + 6	8	0	0	A
161	Reactor Bldg (Unit 2)	180°-225°	Bel. El.586 + 6	5	0	0	A
162	Reactor Bldg (Unit 2)	315°-0°	Bel. El.586 + 6	5	0	0	A
163	Reactor Bldg (Unit 2)	0°-45°	Bel. El.593 + 2½	13	0	0	A
164	Reactor Bldg (Unit 2)	45°-90°	Bel. El.593 + 2½	17	0	0	A
165	Reactor Bldg (Unit 2)	90°-135°	Bel. El.593 + 2½	13	0	0	A
166	Reactor Bldg (Unit 2)	135°-180°	Bel. El.593 + 2½	10	0	0	A
167	Reactor Bldg (Unit 2)	180°-225°	Bel. El.593 + 2½	2	0	0	A
168	Reactor Bldg (Unit 2)	315°-0°	Bel. El.593 + 2½	7	0	0	A
169	RCP-1A	Reactor Bldg.	El.593 + 2½	0	0	1	A
170	RCP-1B	Reactor Bldg.	El.593 + 2½	0	0	1	A
171	RCP-1C	Reactor Bldg.	El.593 + 2½	0	0	1	A
172	RCP-1D	Reactor Bldg.	El.593 + 2½	0	0	1	A
173	RCP 2A	45°	Bel. El.593 + 2½	0	0	1	A
174	RCP 2B	135°	Bel. El.593 + 2½	0	0	1	A
175	RCP 2C	225°	Bel. El.593 + 2½	0	0	1	A
176	RCP 2D	315°	Bel. El.593 + 2½	0	0	1	A
177	Filter Bed Unit 1B	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A
178	Filter Bed Unit 1A	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A
179	Filter Bed Unit 2A	Reactor Bldg.	Bel. El.565 + 3	2	0	2	A

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TABLE 4.3-8 (Continued)

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TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exist:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure (Alarm only), or

c. Instrument indicates a downscale failure (Alarm only).

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With more than one PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour. (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

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LIMITING CONDITIONS FOR OPERATION

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SURVEILLANCE REQUIREMENTS

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path which is accessible during plant operations is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months by verifying that each valve (manual, power-operated, or automatic) in the flow path which is inaccessible during plant operations is in its correct position, and

d. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2) By a visual inspection of each Sprinkler System starting at the system isolation valve to verify the system's integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

OCT 7 1983

TABLE 3.7-3 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK #
59, DD	574+0	1RF480
60, AA	574+0	1RF481
49, BB-CC	577+0	1RF490
45, BB	577+0	1RF491
55, DD	574+0	1RF492
54, AA	574+0	1RF493
63, AA	577+0	1RF993
51, AA	577+0	1RF998
62, NN	594+0	1RF205
57, MM	594+0	1RF222
63, JJ	594+0	1RF231
57, HH	594+0	1RF245
57, EE	594+0	1RF253
51, JJ	594+0	1RF259
53, NN	594+0	1RF275
64, BB	594+0	1RF984
50, BB	594+0	1RF985
51, JJ	605+10	1RF265
63, JJ	605+10	1RF233
63-64, MM	631+6	1RF483
50-51, MM	631+6	1RF495
2. Fuel Pools		
48, TT-UU 65	605+10	1RF208
48, TT-UU	605+10	1RF276
~~50-51~~, MM 63	605+10	1RF483
50-51, MM	605+10	1RF822
3. Nuclear Service Water Pump Structure		
East Section	600+0	1RF939
West Section	600+0	1RF940

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3/4.7.13 STANDBY SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 The Standby Shutdown System (SSS) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION: (Units 1 and 2)

a. With the Standby Shutdown System inoperable, restore the inoperable equipment to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

b. With the total leakage from UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE and reactor coolant pump seal leakage greater than 26 gpm, declare the Standby Makeup Pump inoperable and take ACTION a., above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.13.1 The Standby Shutdown System diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel level in the fuel storage tank is greater than or equal to 67 inches, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes at greater than or equal to 700 kW.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM-D975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.13.2 The Standby Shutdown System diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery is above the plates; and

2) The overall battery voltage is greater than or equal to 24 volts.

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TABLE 3.8-1a (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXR-F01B Primary Bkr Backup Fuse	Incore Instrument Room Ventilation Unit 1B Fan Motor
1MXR-F02B Primary Bkr Backup Fuse	Control Rod Drive Vent Fan Motor 1D
1MXR-F03A Primary Bkr Backup Fuse	Lower Containment Ventilation Unit 1D Fan Motor
1MXR-F04C Primary Bkr Backup Fuse	Upper Containment Ventilation Unit 1D Fan Motor
1MXY-F02A Primary Bkr Backup Fuse	NC Pump 1A Oil Lift Pump Motor 1
1MXY-F02B Primary Bkr Backup Fuse	NC Pump 1D Oil Lift Pump Motor 1
1MXY-F03A Primary Bkr Backup Fuse	Reactor Coolant Drain Tank Pump Motor 1A
1MXY-F03D Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Pump Motor 1A
1MXY-F05A Primary Bkr Backup Fuse	Lighting Transformer 1LR8
1MXY-F05B Primary Bkr Backup Fuse	Lighting Transformer 1LR11

1MXY-F02C
Primary Bkr
Backup Fuse

Reactor Building Lower Containment
Welding Machine Receptacle 1RCPL0185

OCT 7 1985

TABLE 3.8-1a (Continued)

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UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
5. 120 VAC Panelboards (Continued)	
1KPN-2 Primary Bkr Backup Fuse	NC Pump Motor 1D Space Heater
1KPN-7-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 1B Fan Motor Space Heater
1KPN-8-1 Primary Bkr Backup Fuse	Lower Containment Vent Unit 1D Fan Motor Space Heater
1KPN-11 Primary Bkr Backup Fuse	Misc Control Power for 1ATC 24
6. DC Welding Circuits	
1EQCB0001 Primary Bkr - AA Backup Bkr - AB	Lower Containment DC Welding Circuit
1EQCB0002 Primary Bkr - AA Backup Bkr - AB	Upper Containment DC Welding Circuit

OCT 7 1985

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TABLE 3.8-1b (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXS-F05A Primary Bkr Backup Fuse	S/G 2A Blowdown Inside Cont Isol Vlv 2BB56A
2EMXS-F05B Primary Bkr Backup Fuse	S/G 2C Blowdown Inside Cont Isol Vlv 2BB60A
2EMXS-F05C Primary Bkr Backup Fuse	Pzr Liquid Sample Line Inside Cont Isol Vlv 2NM3A
2EMXS-F06A Primary Bkr Backup Fuse	Pzr Steam Sample Line Inside Cont Isol Vlv 2NM6A
2EMXS-F06B Primary Bkr Backup Fuse	NC Hot Leg A Sample Line Inside Cont Isol Vlv 2NM22A
2EMXS-F06C Primary Bkr Backup Fuse	NC Hot Leg C Sample Line Inside Cont Isol Vlv 2NM25A
2MXM-F01A Primary Bkr Backup Fuse	Reactor Coolant Pump Motor Drain Tank Pump Motor
2MXM-F02A Primary Bkr Backup Fuse	2 NC Pump 1B Oil Lift Pump Motor 1
2MXM-F02B Primary Bkr Backup Fuse	NC Pump 2C Oil Lift Pump Motor 1
2MXM-F03A Primary Bkr Backup Fuse	Ice Condenser Power Transformer ICT2A
2MXM-F03B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 2B6 Fan Motor A & B

OCT 7 1985

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cancel{\times} \cdot\ Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide ($\cancel{s^{-1}}$ sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (~~s~~ Sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

JUL 18 1985
OCT 7 1985

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CHEMICAL TREATMENT PONDS

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each chemical treatment pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases,

Where:

A_j = pond inventory limit for single radionuclide "j", in Curies;

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", microCuries/ml;

V = design volume of liquid and slurry in the pond, in gallons; and

264 = conversion unit, microCuries/Curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed ponds exceeding the above limit, immediately suspend all additions of radioactive material to the pond and initiate corrective action to reduce the contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of resin/water slurry to be transferred to the chemical treatment ponds shall be determined to be within the above limit by analyzing a representative sample of the batch to be transferred to the chemical treatment ponds and shall be limited by the expression:

$$\sum_j \frac{c_j}{C_j} < 0.006$$

Where:

c_j = radioactive resin/water slurry concentration for radionuclide "j" entering the UNRESTRICTED AREA chemical treatment ponds, in microCuries/milliliter; and

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

JUL 18 1985
OCT 7 1985

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate. (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (~~s^{-1}~~ sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (~~s~~ sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

JUL 18 1985
JUL 18 1985
OCT 7 1985

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

(raise)

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy ~~vegetable~~ vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentrations of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide ($\cancel{s^{-1}}$), and sec^{-1}

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting ($\cancel{s}$). sec

Typical values of E, V, Y and Δt should be used in the calculation.

JUL 18 1985
OCT 7 1985

FIRE DETECTION INSTRUMENTATION (Continued)

any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION SYSTEM

The OPERABAILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} ~~μCi/cc~~ μCi/ml are measurable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

Release Reports shall include the following information for each ~~type~~ class of solid waste (as defined by 10CFR Part 61) shipped offsite during the report period:

a. Total Container volume, [redacted]

b. Total Curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source ~~Type~~ of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. [redacted] Type of container (e.g. LSA, Type A, Type B, large quantity), and

f. Solidification agent or absorbent (e.g., cement ~~or other approved agents (media)]~~ urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form on an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation". Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.