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

November 10, 1987

Docket Nos.: 50-413
and 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 32 to Facility Operating License NPF-35 and Amendment No. 23 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 65589/65590)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 32 to Facility Operating License NPF-35 and Amendment No. 23 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated June 12, 1987, and supplemented June 23 and August 12, 1987.

The amendments modify the Technical Specifications to permit operation up to full power with the Upper Head Injection Accumulator System removed.

A copy of the related safety evaluation supporting Amendment No.32 to Facility Operating License NPF-35 and Amendment No. 23 to Facility Operating License NPF-52 is enclosed.

Also enclosed is a copy of the related Federal Register notice.

Sincerely,

Kahtan N. Jabbour

Kahtan Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

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Enclosures:
1. Amendment No. 32 to NPF-35
2. Amendment No. 23 to NPF-52
3. Safety Evaluation
4. F.R. Notice

cc w/encl:
See next page

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Mr. H. B. Tucker
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Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc., (licensees) dated June 12, 1987, and supplemented June 23 and August 12, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 32, and the Environmental Protection Plan

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contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Lawrence P. Crocker

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: November 10, 1987

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, (licensees) dated June 12, 1987, and supplemented June 23 and August 12, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 23 , and the Environmental Protection Plan

contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Lawrence P. Crocker

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: November 10, 1987

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 3-75	3/4 3-76
3/4 4-22	3/4 4-21
3/4 5-1	
3/4 5-1a	
3/4 5-2	
3/4 5-3	
3/4 5-4	
3/4 6-7	
3/4 6-23	
3/4 6-24	
3/4 6-31	
3/4 6-32	
3/4 8-23	
3/4 8-24	
3/4 8-46	3/4 8-45
3/4 8-47	3/4 8-48
B 3/4 5-1	B 3/4 5-2
B 3/4 5-3	

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION			MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
69	Cable Room		CC-59	El.577 + 0	18	0	15	A
71	Elect Pene. Room		CC-51	El.594 + 0	10	0	0	A
72	Control Room		CC-56	El.594 + 0	23	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

*Upon the disconnection of the UHI System from the Reactor Coolant System, these specifications are no longer applicable.

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	DESCRIPTION	LOCATION		MINIMUM INSTRUMENTS OPERABLE* SMOKE	FLAME	HEAT	FUNCTION**
147	Reactor Bldg.	135°-180°	Bel. El.593 + 2½	10	0	0	A
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	0	0	A
152	Reactor Bldg. (Unit 2)	90°-135°	Bel. El.565 + 3	4	0	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

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous radioactivity monitor at least once per 12 hours;

b. Monitoring the containment floor and equipment sumps inventory and discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the reactor head flange leakoff at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
NI59	Accumulator Discharge
NI60	Accumulator Discharge
NI70	Accumulator Discharge
NI71	Accumulator Discharge
NI81	Accumualtor Discharge
NI82	Accumulator Discharge
NI93	Accumulator Discharge
NI94	Accumulator Discharge
NI124	Safety Injection (Hot Leg)
NI125	Residual Heat Removal (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI129	Residual Heat Removal (Hot Leg)
NI134	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI165	Safety Injection/Residual Heat Removal (Cold Leg)
NI167	Safety Injection/Residual Heat Removal (Cold Leg)
NI169	Safety Injection/Residual Heat Removal (Cold Leg)
NI171	Safety Injection/Residual Heat Removal (Cold Leg)
NI175	Safety Injection/Residual Heat Removal (Cold Leg)
NI176	Safety Injection/Residual Heat Removal (Cold Leg)
NI180	Safety Injection/Residual Heat Removal (Cold Leg)
NI181	Safety Injection/Residual Heat Removal (Cold Leg)
NI248	Upper Head Injection
NI249	Upper Head Injection
NI250	Upper Head Injection
NI251	Upper Head Injection
NI252	Upper Head Injection
NI253	Upper Head Injection

Note 1

Note 1: Upon the disconnection of the UHI System from the Reactor Coolant System, this specification is no longer applicable.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7853 and 8171 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 385 and 481 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. (UHI System Operable)

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

   1) The volume weighted average boron concentration of the three limiting accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

   2) The volume weighted average boron concentration of the three limiting accumulators less than 1900 ppm but greater than 1500 ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

*Pressurizer pressure above 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

3) The volume weighted average boron concentration of the three limiting accumulators 1500 ppm or less, return the volume weighted average boron concentration of the three limiting accumulators to greater than 1500 ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1.2 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7704 and 8004 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 678 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. (UHI physically disconnected; Cold Leg Accumulators and discharge paths suitably modified)

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.2.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:**

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

4.5.1.1.2.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

** This surveillance need not be performed until prior to entering HOT STANDBY following the Unit 1 refueling.

UPPER HEAD INJECTION

(Deleted upon the physical disconnection of the UHI System from the Reactor Coolant System)

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M386	Containment Air Release	Auxiliary Building	Type C
M204	Containment Air Addition	Auxiliary Building	Type C
M316	Int. Fire Protection Header - Hose Racks	Auxiliary Building	Type C
M337	Demineralized Water	Auxiliary Building	Type C
M220	Instrument Air	Auxiliary Building	Type C
M219	Station Air	Auxiliary Building	Type C
M215	Breathing Air	Auxiliary Building	Type C
M329	Reactor Coolant Pump Motor Oil Fill	Auxiliary Building	Type C
M361	Int. Fire Protection Header - Sprinklers	Auxiliary Building	Type C
M119	Containment Purge Exhaust	Auxiliary Building	Type C
M331	Nitrogen Supply to Cold Leg Accumulators	Auxiliary Building	Type C
M322	Safety Injection Test Line	Auxiliary Building	Type C
M454	UHI Test Line	Auxiliary Building	Type C } Note 1
M328*	Component Cooling to Reactor Vessel Support and RCP Coolers	Auxiliary Building	Type C

*Not applicable for Unit 1 until after the first refueling outage.

Note 1: Upon capping of penetrations associated with deletion of UHI, this specification is no longer applicable.

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
KC-305B#	Excess Letdown Hx Supply Containment Isolation (Outside)	≤20
KC-315B#	Excess Letdown Hx Return Header Containment Isolation (Outside)	≤20
KC-320A#	NCDT Hx Supply Hdr Containment Isolation (Outside)	≤20
KC-332B#	NCDT Hx Return Hdr Containment Isolation (Inside)	≤20
KC-333A#	NCDT Hx Return Hdr Containment Isolation (Outside)	≤20
KC-429B	RB Drain Header Inside Containment Isolation	≤10
KC-430A	RB Drain Header Outside Containment Isolation	≤10
NB-260B	Reactor Makeup Water Tank to Flush Header	≤10
NC-53B	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Outside	≤10
NC-54A	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Inside	≤10
NC-56B	RMW Pump Disch Cont Isolation	≤10
NC-195B	NC Pump Motor Oil Containment Isolation Outside	≤10
NC-196A	NC Pump Motor Oil Containment Isolation Inside	≤10
NF-228A	Unit 1 Air Handling Units Glycol Supply Containment Isolation Outside	≤10
NF-233B	Unit 1 Air Handling Units Glycol Return Containment Isolation Inside	≤10
NF-234A	Unit 1 Air Handling Units Glycol Return Containment Isolation Outside	≤10
NI-47A	Accumulator N_2 Supply Outside Containment Isolation	≤10
NI-95A	Test Hdr Inside Containment Isolation	≤10
NI-96B	Test Hdr Outside Containment Isolation	≤10
NI-120B	Safety Injection Pump to Accumulator Fill Line Isolation	≤10
NI-122B#	Hot Leg Injection Check 1NI124, 1NI128 Test Isolation	≤10
NI-154B#	Hot Leg Recirculation Check 1NI125, 1NI129 Test Isolation	≤10
NI-255B	UHI Check Valve Test Line Isolation	≤10 } Note 1
NI-258A	UHI Check Valve Test Line Isolation	≤10 } Note 1
NI-264B	UHI Check Valve Test Line Outside Containment Isolation	≤10 } Note 1

Note 1: Upon capping of penetrations associated with deletion of UHI, these specifications are no longer applicable.

TABLE 3.6-2a (Continued)

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NI-266A	UHI Check Valve Test Line Inside Containment Isolation	≤10 Note 1
NI-267A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-153A#	Hot Leg Injection Check NI156, NI159 Test Isolation	≤10
NM-3A	Pressurizer Liquid Sample Line Inside Containment Isolation	≤10
NM-6A	Pressurizer Steam Sample Line Inside Containment Isolation	≤10
NM-7B	Pressurizer Sample Header Outside Containment Isolation	≤10
NM-22A	NC Hot Leg A Sample Line Inside Containment Isolation	≤10
NM-25A	NC Hot Leg C Sample Line Inside Containment Isolation	≤10
NM-26B	NC Hot Leg Sample Hdr Outside Containment Isolation	≤10
NM-72B	NI Accumulator 1A Sample Line Inside Containment Isolation	≤10
NM-75B	NI Accumulator 1B Sample Line Inside Containment Isolation	≤10
NM-78B	NI Accumulator 1C Sample Line Inside Containment Isolation	≤10
NM-81B	NI Accumulator 1D Sample Line Inside Containment Isolation	≤10
NM-82A	NI Accumulator Sample Hdr Outside Containment Isolation	≤10
NM-187A#	SG 1A Upper Shell Sample Containment Isolation Inside	≤10
NM-190A#	SG 1A Blowdown Line Sample Containment Isolation Inside	≤10
NM-191B#	SG 1A Sample Hdr Containment Isolation Outside	≤10
NM-197B#	SG 1B Upper Shell Sample Containment Isolation Inside	≤10
NM-200B#	SG 1B Blowdown Line Sample Containment Isolation Inside	≤10
NM-201A#	SG 1B Sample Hdr Containment Isolation Outside	≤10
NM-207A#	SG 1C Upper Shell Sample Containment Isolation Inside	≤10
NM-210A#	SG 1C Blowdown Line Sample Containment Isolation Inside	≤10
NM-211B#	SG 1C Sample Hdr Containment Isolation Outside	≤10
NM-217B#	SG 1D Upper Shell Sample Containment Isolation Inside	≤10
NM-220B#	SG 1D Blowdown Line Sample Containment Isolation Inside	≤10
NM-221A#	SG 1D Sample Hdr Containment Isolation Outside	≤10
NV-15B	Letdown Containment Isolation Outside	≤10
NV-89A	NC Pumps Seal Return Containment Isolation Inside	≤10
NV-91B	NC Pumps Seal Return Containment Isolation Outside	≤10
NV-314B#	Charging Line Containment Isolation Outside	≤10

Note 1: Upon capping of penetrations associated with deletion of UHI, these specifications are no longer applicable.

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
KC-305B#	Excess Letdown Hx Supply Containment Isolation (Outside)	≤20
KC-315B#	Excess Letdown Hx Return Header Containment Isolation (Outside)	≤20
KC-320A#	NCDT Hx Supply Hdr Containment Isolation (Outside)	≤20
KC-332B#	NCDT Hx Return Hdr Containment Isolation (Inside)	≤20
KC-333A#	NCDT Hx Return Hdr Containment Isolation (Outside)	≤20
KC-429B	RB Drain Header Inside Containment Isolation	≤10
KC-430A	RB Drain Header Outside Containment Isolation	≤10
NB-260B	Reactor Makeup Water Tank to Flush Header	≤10
NC-53B	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Outside	≤10
NC-54A	Nitrogen to Pressurizer Relief Tank #1 Containment Isolation Inside	≤10
NC-56B	RMW Pump Disch Cont Isolation	≤10
NC-195B	NC Pump Motor Oil Containment Isolation Outside	≤10
NC-196A	NC Pump Motor Oil Containment Isolation Inside	≤10
NF-228A	Unit 2 Air Handling Units Glycol Supply Containment Isolation Outside	≤10
NF-233B	Unit 2 Air Handling Units Glycol Return Containment Isolation Inside	≤10
NF-234A	Unit 2 Air Handling Units Glycol Return Containment Isolation Outside	≤10
NI-47A	Accumulator N_2 Supply Outside Containment Isolation	≤10
NI-95A	Test Hdr Inside Containment Isolation	≤10
NI-96B	Test Hdr Outside Containment Isolation	≤10
NI-120B	Safety Injection Pump to Accumulator Fill Line Isolation	≤10
NI-122B#	Hot Leg Injection Check 2NI124, 2NI128 Test Isolation	≤10
NI-154B#	Hot Leg Recirculation Check 2NI125, 2NI129 Test Isolation	≤10
NI-255B	UHI Check Valve Test Line Isolation	≤10 } Note 1
NI-258A	UHI Check Valve Test Line Isolation	≤10 } Note 1
NI-264B	UHI Check Valve Test Line Outside Containment Isolation	≤10 } Note 1

Note 1: Upon capping of penetrations associated with deletion of UHI, these specifications are no longer applicable.

TABLE 3.6-2b (Continued)

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation (Continued)		
NI-266A	UHI Check Valve Test Line Inside Containment Isolation	≤10 Note 1
NI-267A	UHI Check Valve Test Line Inside Containment Isolation	≤10
NI-153A#	Hot Leg Injection Check NI156, NI159 Test Isolation	≤10
NM-3A	Pressurizer Liquid Sample Line Inside Containment Isolation	≤10
NM-6A	Pressurizer Steam Sample Line Inside Containment Isolation	≤10
NM-7B	Pressurizer Sample Header Outside Containment Isolation	≤10
NM-22A	NC Hot Leg A Sample Line Inside Containment Isolation	≤10
NM-25A	NC Hot Leg C Sample Line Inside Containment Isolation	≤10
NM-26B	NC Hot Leg Sample Hdr Outside Containment Isolation	≤10
NM-72B	NI Accumulator 2A Sample Line Inside Containment Isolation	≤10
NM-75B	NI Accumulator 2B Sample Line Inside Containment Isolation	≤10
NM-78B	NI Accumulator 2C Sample Line Inside Containment Isolation	≤10
NM-81B	NI Accumulator 2D Sample Line Inside Containment Isolation	≤10
NM-82A	NI Accumulator Sample Hdr Outside Containment Isolation	≤10
NM-187A#	SG 2A Upper Shell Sample Containment Isolation Inside	≤10
NM-190A#	SG 2A Blowdown Line Sample Containment Isolation Inside	≤10
NM-191B#	SG 2A Sample Hdr Containment Isolation Outside	≤10
NM-197B#	SG 2B Upper Shell Sample Containment Isolation Inside	≤10
NM-200B#	SG 2B Blowdown Line Sample Containment Isolation Inside	≤10
NM-201A#	SG 2B Sample Hdr Containment Isolation Outside	≤10
NM-207A#	SG 2C Upper Shell Sample Containment Isolation Inside	≤10
NM-210A#	SG 2C Blowdown Line Sample Containment Isolation Inside	≤10
NM-211B#	SG 2C Sample Hdr Containment Isolation Outside	≤10
NM-217B#	SG 2D Upper Shell Sample Containment Isolation Inside	≤10
NM-220B#	SG 2D Blowdown Line Sample Containment Isolation Inside	≤10
NM-221A#	SG 2D Sample Hdr Containment Isolation Outside	≤10
NV-15B	Letdown Containment Isolation Outside	≤10
NV-89A	NC Pumps Seal Return Containment Isolation Inside	≤10
NV-91B	NC Pumps Seal Return Containment Isolation Outside	≤10
NV-314B#	Charging Line Containment Isolation Outside	≤10

Note 1: Upon capping of penetrations associated with deletion of UHI, these specifications are no longer applicable.

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED	
2. 600 VAC MCC (Continued)		
1EMXD-F02C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC31B	
1EMXD-F05A Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC35B	
1EMXD-F05B Primary Bkr Backup Fuse	Rx Bldg Drain Hdr Inside Cont Isol Vlv 1KC429B	
1EMXD-F05C Primary Bkr Backup Fuse	NCDT Hx Clng Water Return Inside Isol Vlv 1KC332B	
1EMXD-F06A Primary Bkr Backup Fuse	NC Pump 1B Thermal Barrier Outlet Isol Vlv 1KC364B	
1EMXD-F06B Primary Bkr Backup Fuse	NC Pumps Rtn Hdr Inside Cont Isol Vlv 1KC424B	
1EMXK-F01A Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 1NI266A	Note 1
1EMXK-F01B Primary Bkr Backup Fuse	Upper Cont Vent Units Return Cont Isol Vlv 1RN429A	
1EMXK-F01C Primary Bkr Backup Fuse	Backup N_2 to PORV 1NC34A From Accum Tnk 1A Vlv 1NI438A	
1EMXK-F02A Primary Bkr Backup Fuse	NC Pump 1A Thermal Barrier Outlet Isol Vlv 1KC394A	
1EMXK-F02B Primary Bkr Backup Fuse	Lower Cont Vent Units Return Cont Isol Vlv 1RN484A	

Note 1: Upon removal of cable from power source associated with the deletion of UHI, this specification is no longer applicable.

TABLE 3.8-1A (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED	
2. 600 VAC MCC (Continued)		
1EMXK-F02C Primary Bkr Backup Fuse	NV Supply to Pressurizer Vlv 1NV037A	
1EMXK-F03A Primary Bkr Backup Fuse	S/G C Blowdown Line Sample Inside Cont Isol Vlv 1NM210A	
1EMXK-F04A Primary Bkr Backup Fuse	S/G A Upper Shell Sample Inside Cont Isol Vlv 1NM187A	
1EMXK-F04B Primary Bkr Backup Fuse	S/G A Blowdown Line Sample Inside Cont Isol Vlv 1NM190A	
1EMXK-F04C Primary Bkr Backup Fuse	S/G C Upper Shell Sample Inside Cont Isol Vlv 1NM207A	
1EMXK-F06A Primary Bkr Backup Fuse	Hydrogen Skimmer Fan 1A Inlet Vlv 1VX1A	
1EMXK-F07C Primary Bkr Backup Fuse	Electric Hydrogen Recombiner Power Supply Panel 1A	
1EMXK-F09A Primary Bkr Backup Fuse	Accumulator 1A Discharge Isol Vlv 1NI54A	
1EMXK-F09B Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 1NI267A	Note 1
1EMXK-F09C Primary Bkr Backup Fuse	NC Pump Oil Fill Header Cont Isol Vlv 1NC196A	

Note 1: Upon removal of cable from power source associated with the deletion of UHI, this specification is no longer applicable.

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXC-F05A Primary Bkr Backup Fuse	NCDT Vent Inside Cont Isol Vlv 2WL450A
2EMXC-F05B Primary Bkr Backup Fuse	Cont Sump Pumps Discharge Inside Cont Isol Vlv 2WL825A
2EMXC-F05C Primary Bkr Backup Fuse	Vent Unit Cond Drn Tank Outside Cont Isol Vlv 2WL867A
2EMXC-F06A Primary Bkr Backup Fuse	NCDT Pumps Disch Inside Cont Isol Vlv 2WL805A
2EMXC-F07B Primary Bkr Backup Fuse	Cont H_2 Purge Outlet Cont Isol Vlv 2VY17A
2EMXD-F01A Primary Bkr Backup Fuse	ND Pump 2A Suction From NC Loop B Vlv 2ND1B
2EMXD-F01B Primary Bkr Backup Fuse	Accumulator 2B Discharge Isol Vlv 2NI65B
2EMXD-F01C Primary Bkr Backup Fuse	NI Pump A to Hot Leg Check Vlv Test Isol Vlv 2NI122B
2EMXD-F02A Primary Bkr Backup Fuse	ND Pump 2B Suction from NC Loop C Vlv 2ND36B
2EMXD-F02B Primary Bkr Backup Fuse	ND to Hot Legs Chk 2NI125, 2NI129 Test Isol Vlv 2NI154B

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED	
2. 600 VAC MCC (Continued)		
2EMXD-F02C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 2NC31B	
2EMXD-F05A Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 2NC35B	
2EMXD-F05B Primary Bkr Backup Fuse	Rx Bldg Drain Hdr Inside Cont Isol Vlv 2KC429B	
2EMXD-F05C Primary Bkr Backup Fuse	NCDT Hx Clng Water Return Inside Isol Vlv 2KC332B	
2EMXD-F06A Primary Bkr Backup Fuse	NC Pump 2B Thermal Barrier Outlet Isol Vlv 2KC364B	
2EMXD-F06B Primary Bkr Backup Fuse	NC Pumps Rtn Hdr Inside Cont Isol Vlv 2KC424B	
2EMXK-F01A Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 2NI266A	Note 1
2EMXK-F01B Primary Bkr Backup Fuse	Upper Cont Vent Units Return Cont Isol Vlv 2RN429A	
2EMXK-F01C Primary Bkr Backup Fuse	Backup N_2 to PORV 2NC34A From Accum Tnk 2A Vlv 2NI438A	
2EMXK-F02A Primary Bkr Backup Fuse	NC Pump 2A Thermal Barrier Outlet Isol Vlv 2KC394A	
2EMXK-F02B Primary Bkr Backup Fuse	Lower Cont Vent Units Return Cont Isol Vlv 2RN484A	

Note 1: Upon removal of cable from power source associated with the deletion of UHI, this specification is no longer applicable.

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED	
2. 600 VAC MCC (Continued)		
2EMXK-F02C Primary Bkr Backup Fuse	NV Supply to Pressurizer Vlv 2NV037A	
2EMXK-F03A Primary Bkr Backup Fuse	S/G C Blowdown Line Sample Inside Cont Isol Vlv 2NM210A	
2EMXK-F04A Primary Bkr Backup Fuse	S/G A Upper Shell Sample Inside Cont Isol Vlv 2NM187A	
2EMXK-F04B Primary Bkr Backup Fuse	S/G A Blowdown Line Sample Inside Cont Isol Vlv 2NM190A	
2EMXK-F04C Primary Bkr Backup Fuse	S/G C Upper Shell Sample Inside Cont Isol Vlv 2NM207A	
2EMXK-F06A Primary Bkr Backup Fuse	Hydrogen Skimmer Fan 2A Inlet Vlv 2VX1A	
2EMXK-F07C Primary Bkr Backup Fuse	Electric Hydrogen Recombiner Power Supply Panel 2A	
2EMXK-F09A Primary Bkr Backup Fuse	Accumulator 2A Discharge Isol Vlv 2NI54A	
2EMXK-F09B Primary Bkr Backup Fuse	UHI Check Vlv Test Line Inside Cont Isol Vlv 2NI267A	Note 1
2EMXK-F09C Primary Bkr Backup Fuse	NC Pump Oil Fill Header Cont Isol Vlv 2NC196A	

Note 1: Upon removal of cable from power source associated with the deletion of UHI, this specification is no longer applicable.

TABLE 3.8-1B (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
2EMXK-F10A Primary Bkr Backup Fuse	Containment Air Return Damper 2ARF-D-2
2EMXK-F10B Primary Bkr Backup Fuse	VQ Fans Suction From Containment Isol Vlv 2VQ2A
2EMXK-F10C Primary Bkr Backup Fuse	Cont Air Addition Containment Isol Vlv 2VQ16A
2EMXK-F11A Primary Bkr Backup Fuse	Containment Air Return Fan Motor 2A
2EMXK-F11B Primary Bkr Backup Fuse	Hydrogen Skimmer Fan Motor 2A
2EMXL-F01B Primary Bkr Backup Fuse	Trn B Alternate Power to ND Letdn Vlv 2ND37A
2EMXL-F01C Primary Bkr Backup Fuse	NI Accum D Sample Line Inside Cont Isol Vlv 2NM81B
2EMXL-F02A Primary Bkr Backup Fuse	NC Pump 2D Thermal Barrier Outlet Isol Vlv 2KC413B
2EMXL-F02B Primary Bkr Backup Fuse	Air Handling units Glycol Return Cont. Isol Vlv 2NF233B
2EMXL-F02C Primary Bkr Backup Fuse	NI Accum C Sample Line Inside Cont Isol Vlv 2NM78B
2EMXL-F03A Primary Bkr Backup Fuse	S/G D Blowdown Sample Line Inside Cont Isol Vlv 2NM220B

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs from the cold leg injection accumulators and directly into the reactor vessel from the upper head injection accumulators in the event the Reactor Coolant System pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The allowed down time for the accumulators are variable based upon boron concentration to ensure that the reactor is shutdown following a LOCA and that any problems are corrected in a timely manner. Subcriticality is assured when boron concentration is above 1500 ppm, so additional down time is allowed when concentration is above 1500 ppm. A concentration of less than 1900 ppm in any single accumulator or as a volume weighted average may be indicative of a problem, such as valve leakage, but since reactor shutdown is assured, additional time is allowed to restore boron concentration in the accumulators.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The original licensing basis for the Catawba Nuclear Station assumes both the UHI System and Cold Leg Accumulators function to mitigate postulated accidents. Subsequent analyses have demonstrated that the UHI System is not required provided minor changes to the Cold Leg Accumulator parameters and discharge paths are implemented. Accordingly, Specification 3/4.5.1 has been modified to address two possible plant configurations:

(1) UHI Operable (Original plant configuration)

(2) UHI Disconnected (UHI penetrations to the Reactor Vessel are cut and capped, Cold Leg Accumulator level and cover pressure changed, Cold Leg Accumulator flow restricting orifice replaced)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the coolant temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE centrifugal charging pump to be inoperable below 285°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the refueling water storage tank and the Reactor Coolant System water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

REFUELING WATER STORAGE TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 8.0 and 9.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NPF-52

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DUKE POWER COMPANY, ET AL.

I. INTRODUCTION

By letter dated June 12, 1987, (Ref. 1) and supplemented June 23 and August 12, 1987, (Refs. 2 and 3), Duke Power Company, et al. (the licensee), proposed changes to the Technical Specifications to request operation up to full power with the Upper Head Injection (UHI) System removed. The UHI system was designed for use in Catawba to enhance core cooling during the blowdown phase of the loss-of-coolant accidents (LOCA). The removal of the UHI system is to increase plant operational flexibility. The proposed changes include: (1) deletion of Technical Specifications requiring UHI system surveillance, maintenance and leakage verification, and (2) modification of the Technical Specifications to reflect removal of the UHI system. Similar changes have previously been approved by NRC for incorporation into the McGuire Technical Specifications (Ref. 4). The staff has reviewed the proposed Technical Specification changes and the supporting analytical results, and has prepared the following evaluation.

II. EVALUATION

1. Peak Containment Pressure and Temperature

The peak containment pressure originally calculated in the Final Safety Analysis Report (FSAR) was determined for the design basis LOCA conservatively assuming there was no UHI system in operation. Therefore, the current calculated peak pressure in the FSAR is still valid after the UHI system is removed. However, the main steam line break (MSLB) analysis in the FSAR for determining peak containment temperature has been revised due to the concern regarding superheated steam releases described in IE Information Notice 84-90. In the revised analysis, the limiting case MSLB was determined to be a 0.86 Ft^2 break at 102% reactor power. The UHI system would not be actuated for this break because the actuation pressure of 1200 psia in the primary system could not be reached. The 1200 psia actuation pressure can be reached only for breaks at much lower power (less than 30%). It should be noted, however, that for breaks at low power, the integrated blowdown mass and energy, maximum steam enthalpy, and therefore, the containment temperature are less than the limiting case discussed above. The licensee, therefore, concluded that the deletion of the UHI system has no impact on the limiting case peak containment temperature. Based on the above, the staff concurs with the licensee that deletion of the UHI system has no effect on the peak containment temperature calculation as long as the limiting case is a MSLB at 102% power.

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2. Minimum Containment Pressure Analysis for Performance Capability Studies of ECCS

Appendix K to 10 CFR 50 requires that the containment pressure used for evaluation of cooling effectiveness during reactor core reflood shall not exceed a pressure calculated conservatively for this purpose. The calculation must include the effect of operation of all installed containment pressure-reducing systems and processes. The corresponding reflood rate in the core will then be reduced because lessened containment pressure reduces the resistance to steam flow in the reactor coolant loops and increases the boiloff rate from the core.

The licensee has performed the containment back pressure calculation for ECCS performance studies using the computer code LOTIC-2. This is consistent with the FSAR. The input of mass and energy release rates corresponding to a double-ended cold leg guillotine break (CO = 0.6) were calculated using the method described in the revised FSAR Section 15.6.5, Loss-of-Coolant Accidents. However, the computer code BASH, instead of the previously used code WREFLOOD, was employed to compute the reflood transient.

The staff has revised the licensee's input data including initial containment conditions, containment net free volume and passive heat sinks. Some of these data were revised from the FSAR to reflect the as-built plant configuration. The staff finds these data acceptable. Based on the above, the staff finds the minimum containment pressure calculated for ECCS performance to be acceptable.

3. Large Break Loss-of-Coolant-Accident (LOCA) Analysis

The licensee provided the results of a large break LOCA analysis supporting the request for removal of the UHI system. In the licensee's submittal only double-ended cold leg guillotine (DECLG) breaks were analyzed since they were identified previously as limiting cases that result in the highest peak cladding temperature. The DECLG break analysis was performed with a total peaking factor of 2.32, 102% of the core power of 3411 Mwt and an assumed loss of offsite power at the beginning of the accident. A sensitivity study of DECLG break sizes on the effect of the peak cladding temperature was performed by use of Moody discharge coefficients of 1.0, 0.8 and 0.6. The results showed that the DECLG break with a discharge coefficient of 0.8 is the worst large break LOCA case, resulting in a peak cladding temperature of 1704°F. The analysis was performed with a modified version of the 1981 Westinghouse ECCS evaluation model (Ref. 5). This evaluation model used the revised PAD fuel thermal safety model (Ref. 6) for the calculation of the initial fuel conditions; the SATAN-VI code (Ref. 7) for the transient thermal hydraulic calculation during blowdown period; the WREFLOOD (Ref. 8) and BASH (Refs. 9, 10 and 11) codes for the calculation of the refill and reflood transient periods; the LOCBART (Refs. 9 and 12) for the calculation of the peak cladding temperature; and the LOTIC code (Ref. 13) for the calculation of the ice condenser containment pressure transient.

The staff has reviewed the analysis. The staff found that the approved analytical methods and computer codes (including the BASH modifications in Reference 11, which has been approved for the plant specific application) were used and the results showed that the peak cladding temperature, metal-water reaction and clad oxidation are within the acceptance criteria imposed in 10 CFR 50.46 for LOCA analysis.

4. Small Break LOCA Evaluation

The small break LOCA analysis was performed with the approved computer codes, i.e., (1) the NOTRUMP (Refs 14 and 15) code for the calculation of the transient depressurization of the reactor coolant system, core power, water-steam mixture height and steam flow past the uncovered portion of the core, and (2) the LOCTA-IV (Ref. 12) code for the peak cladding temperature analysis. The analysis was done assuming 102% of the core power of 3411 Mwt, and a total peaking factor of 2.32. Various break sizes were performed and the results showed that the worst break size is a 4-inch diameter break which results in the highest peak cladding temperature of 1304°F, well below the acceptance criteria of 2200°F. The staff concludes that the small break LOCA analysis is acceptable since the approved method was used to show the analytical results to be within the acceptance criteria of 10 CFR 50.46.

5. Transient Evaluation

The licensee used the approved LOFTRAN code (Ref. 16) to reanalyze two plant transients which were (1) the inadvertent opening of a steam generator relief or safety valve and (2) the steam line break. These two transients were reanalyzed because they were the only transients which were predicted to depressurize the primary system sufficiently to actuate the UHI system. An analysis to determine whether departure from nucleate boiling (DNB) occurred was performed for both transients and the results confirmed that no DNB occurred and thus assured no fuel damage resulting from the transients.

The staff concludes that the licensee's transient analysis is adequate and acceptable since the approved method was used and the results were demonstrated to be acceptable since specified acceptable fuel design limits were not exceeded.

6. Occupational Radiation Exposure

The licensee has provided an estimate of the occupational radiation exposure for the UHI deletion modification. The estimate is based on anticipated stay times for each major subtask and estimated dose rates. This estimated total dose is 80 person-rem per unit, which is a small fraction of the 1986 annual average PWR dose of 392 person-rem per unit. The licensee is required to maintain occupational radiation exposures within regulatory limits. Furthermore, the licensee has committed to and is implementing a program for maintaining occupational radiation exposures as low as is reasonably achievable. Therefore, the projected occupational radiation exposures associated with the UHI deletion modification are acceptable.

III. Technical Specification Changes

The proposed changes to the Technical Specifications (TS) associated with the removal of the UHI system are discussed below.

(1) TS 3/4.3.3.8 - Fire Detection Instrumentation

The requirements of TS of the fire detection instruments in the UHI building are proposed to be deleted from the previous Technical Specifications because fire protection for the UHI system is no longer needed.

(2) TS 3/4.4.6 - Reactor Coolant System Leakage

The requirements of leak verification for the UHI system related piping and valves are deleted since equipment is no longer part of the reactor coolant system.

(3) TS 3/4.5.1.1 - ECCS, Cold Leg Injection

The proposed change involves (1) the decrease of the operable range of water volume from 7853 and 8171 gallons to 7704 and 8004 gallons, and increase of the operable range of the nitrogen cover pressure from 385 and 481 psia to 585 and 678 psia for the ECCS cold leg accumulators. The changes are consistent with the LOCA analysis supporting the request for removal of the UHI system.

(4) TS 3/4.5.1.2 - ECCS, Upper Head Injection

The Specifications associated with the maintenance of the UHI system are deleted.

(5) TS 3/4.6.1 and 3/4.6.3 - Containment Penetration and Valves

The proposed changes would revise: (1) Table 3.6-1, secondary containment bypass leakage paths to reflect the sealing of the UHI related containment penetration, and (2) Table 3.6-2, containment isolation valves to reflect the removal of containment isolation valves associated with UHI containment penetration. Upon the removal of the UHI system, the UHI penetration will be cut and capped, and the associated piping and valves will be removed. The staff finds capping and sealing the unused containment penetration to be appropriate, and therefore, the associated TS changes are acceptable.

(6) TS 3/4.8.4 - Electrical Equipment Protection Devices

Tables 3.8-1A and 3.8-1B will be revised to reflect the deletion of the UHI system and related containment penetration conductor overcurrent protective devices.

The staff has reviewed the proposed Technical Specification changes and found them to be acceptable because they are consistent with the assumptions used in revised safety analyses which demonstrate that the appropriate acceptance criteria are satisfied without taking credit for the UHI system.

IV. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that issuance of the amendments will have no significant impact on the environment (52 FR 39295).

V. CONCLUSION

Notice of opportunity for a prior hearing was published in the Federal Register on August 27, 1987 (52 FR 32365). No requests for a hearing were received. The Commission's Environmental Assessment and Finding of No Significant Impact was published on October 21, 1987 (52 FR 39295). We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

VI. REFERENCES

1. Letter, with five attachments, from H. Tucker (Duke Power Company) to NRC, dated June 12, 1987.

2. Letter, with an attachment, from H. Tucker (Duke Power Company) to NRC, dated June 23, 1987.

3. Letter, with an attachment, from H. Tucker (Duke Power Company) to NRC dated August 12, 1987.

4. Letter from D. Hood (NRC) to H. Tucker (Duke Power Company), dated May 13, 1986.

5. "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A, Rev. 1, February 1982.

6. "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2, October 1982.

7. "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant," WCAP-8170, June 1974.

8. "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD)," WCAP-8170, June 1974.

9. Letter from C. E. Rossi (NRC) to E. P. Rahe, Jr. (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-10266, the 1981 Version of the Westinghouse ECCS Evaluation Model using the BASH Code," November 13, 1986.

10. "BASH An Integrated Core and RCS Reflood Code for Analysis of PWR Loss-of-Coolant Accident," WCAP-10266, 1984.

11. "BASH Methodology Enhancements," WCAP-10266, Addendum 2, March 1987.

12. "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June 1974.

13. "Long-Term Ice Condenser Containment LOTIC Code Supplement 1," WCAP-8355 (Supplement 1), June 1974.

14. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10081, August 1985.

15. "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10080, August 1985.

16. "LOFTRAN Code Description," WCAP-7907, April 1984.

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John D. Buchanan, PRPB

Dated: November 10, 1987

7590-01

U.S. NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY, ET AL.

DOCKET NOS. 50-413 AND 50-414

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 32 to Facility Operating License No. NPF-35, and Amendment No. 23 to Facility Operating License NPF-52, issued to Duke Power Company, et al., (the licensee), which revised the Technical Specifications for operation of the Catawba Nuclear Station, Units 1 and 2, (the facility) located in York County, South Carolina. The amendments were effective as of the date of issuance.

The amendments modified the Technical Specifications to permit operation up to full power with the Upper Head Injection Accumulator System removed.

The application for the amendments 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 amendments.

Notice of Consideration of Issuance of Amendments and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on August 27, 1987 (52 FR 32365). No request for a hearing or petition for leave to intervene was filed following this notice.

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The Commission has prepared an Environmental Assessment related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which was been predicted and described in the Commission's Final Environmental Statement for the facility dated January 1983.

For further details with respect to the action see (1) the application for amendments dated June 12, 1987, and supplemented June 23 and August 12, 1987, (2) Amendment No. 32 to License No. NPF-35, (3) Amendment No. 23 to License No. NPF-52, and (4) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., and at the York County Library, 138 East Black Street, Rock Hill, South Carolina 29730. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Bethesda, Maryland this 10th day of November 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour

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