

April 3, 1997

Mr. William R. McCollum
Site Vice President
Catawba Nuclear Station
Duke Power Company
4800 Concord Road
York, South Carolina 29745-9635

Distribution C.Castor,RII
Docket File ACRS T-2 E26
PUBLIC OGC
PDII-2 RF G.Hill(4)
S.Varga J.Johnson,RII
C.Grimes

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2
(TAC NOS. M97538 AND M97539)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 158 to Facility Operating License NPF-35 and Amendment No. 150 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments are in response to your application dated January 3, 1997, as supplemented March 20, 1997.

The amendments revise Technical Specification Tables 3.3-2, 3.3-4, 3.3-5, 4.3-2 and Bases Sections 3/4.3.1 and 3/4.3.2 to eliminate the safety injection signal on low steam line pressure.

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

Sincerely,
ORIGINAL SIGNED BY:

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:

1. Amendment No. 158 to NPF-35
2. Amendment No. 150 to NPF-52
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\CATAWBA\CAT97538.AMD

OFFICE	PDII-2/PM	PDII-2/LR	OGC	PDII-2/AD	
NAME	P.TAM:cm	L.BERRY	R.Bachmann	H.BERKOW	
DATE	3/31/97	3/28/97	4/2/97	4/3/97	1/1/97
COPY	YES NO	YES NO	YES NO	YES NO	YES NO

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

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Site Vice President
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Sincerely,

A handwritten signature in black ink, reading "Peter S. Tam", is written over the typed name.

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

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3. Safety Evaluation

cc w/encl: See next page

Catawba Nuclear Station
Units 1 and 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 158
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 January 3, 1997, as supplemented by letter dated March 20, 1997, 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, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

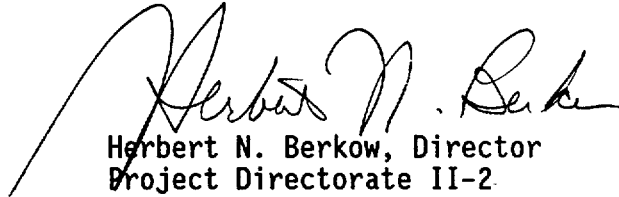
2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 158, which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented before startup from the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: April 3, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 158

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

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.

Remove

3/4 3-15

3/4 3-29

3/4 3-40

3/4 3-44

B 3/4 3-3

Insert

3/4 3-15

3/4 3-29

3/4 3-40

3/4 3-44

B 3/4 3-3

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High	3	2	2	1, 2, 3	15
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3#	19

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High	≤ 1.2 psig	≤ 1.4 psig
d. Pressurizer Pressure-Low	≥ 1845 psig	≥ 1839 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATION SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
4. Steam Line Pressure-Low	
Steam Line Isolation	≤ 10
5. Containment Pressure-High-High	
a. Containment Spray	≤ 45
b. Phase "B" Isolation	≤ 65 ⁽³⁾ /76 ⁽⁴⁾
Nuclear Service Water Operation	N.A.
c. Steam Line Isolation	≤ 10
d. Containment Air Return and Hydrogen Skimmer Operation	≤ 600
6. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	≤ 10

TABLE 4.3-2
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge and Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generators Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High	S	R	Q	N.A.	N.A.	N.A	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	N.A	N.A.	1, 2, 3

INSTRUMENTATION

BASES

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 Defeats the manual block of Safety Injection actuation on low pressurizer pressure and defeats steam line isolation on negative steam line pressure rate. Defeats the manual block of the motor-driven auxiliary feedwater pumps on trip of main feedwater pumps and low-low steam generator water level.

P-12 On decreasing reactor coolant loop temperature, P-12 automatically blocks steam dump and allows manual bypass of steam dump block for the cooldown valves only. On increasing reactor coolant loop temperature, P-12 automatically defeats the manual bypass of the steam dump block.

P-14 On increasing steam generator level, P-14 automatically trips all feedwater isolation valves, pumps and turbine and inhibits feedwater control valve modulation.

Surveillances for the Reactor Trip Bypass Breakers are included in response to the NRC's Generic Letter 85-09, dated May 23, 1985.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. The radiation monitors send actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems. Some of the final actuations are dependent on plant condition in addition to the actuation signals from the radiation monitors.



**UNITED STATES
NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

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. 150
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 January 3, 1997, as supplemented by letter dated March 20, 1997, 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, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment 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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 150, which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented before startup from the current refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION



Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: April 3, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 150

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.

Remove

3/4 3-15
3/4 3-29
3/4 3-40
3/4 3-44

B 3/4 3-3

Insert

3/4 3-15
3/4 3-29
3/4 3-40
3/4 3-44

B 3/4 3-3

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ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
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c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-5 (Continued)

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c. Containment Pressure-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NPF-35
AND AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated January 3, 1997, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). Specifically, the licensee proposed to (1) revise the Technical Specifications, Table 3.3-2, 3.3-4, 3.3-5, 4.3-2 and Bases Section 3/4.3.1 and 3/4.3.2 to eliminate the safety injection signal on low steam line pressure. The initial request was supplemented by a letter dated March 20, 1997, responding to the staff's request for additional information dated March 18, 1997.

2.0 DISCUSSION AND EVALUATION

The safety injection (SI) system is designed to provide borated makeup water during loss of coolant accidents as well as cooldown accidents such as steamline breaks. It is the licensee's goal to minimize unnecessary actuation of the SI system since the introduction of cold water into the primary coolant system can lead to an increase of reactor thermal power, resulting in a thermal transient and pressurization of the reactor coolant system. An added concern to the licensee is that the unnecessary addition of highly borated water into the core could create challenges to plant safety equipment.

The licensee has determined that removing the SI signal on low steamline pressure will limit the number of unnecessary SI actuations. To determine that the removal of the SI actuation on low steamline pressure would not adversely impact the safe operation of the plant, the licensee used staff-approved methodologies to evaluate each of the Updated Final Safety Analysis Report (UFSAR), Chapter 15 transient analyses.

The results of the evaluation determined that the transient analyses fall into three categories: (1) transients which do not involve automatic SI actuation, (2) transients with automatic SI actuation, but initiated by a signal other than low steamline pressure, and (3) transients which involve SI actuation on low steamline pressure. Those transients that involve a significant decrease in steamline pressure were further evaluated or reanalyzed by the licensee.

The purpose of the steamline break analysis is to demonstrate short-term core cooling capability in the event of a steamline break transient. During the licensing review, a spectrum of break sizes were analyzed and documented in the Final Safety Analysis Report (FSAR) for the steamline break transient to determine the most limiting break size. The licensee reanalyzed this event and found that the limiting break sizes remained the same and are 1.4 ft² and 2.0 ft² for Catawba Units 1 and 2, respectively.

The licensee determined that for the smaller breaks the SI will actuate on low pressurizer pressure before reaching the setpoint for SI actuation on low steamline pressure. On the other hand, for larger breaks (greater than 2.5 ft² for Unit 1 and greater than 1.4 ft² for Unit 2) the SI will reach the low steamline pressure actuation setpoint before reaching the low pressurizer pressure SI actuation setpoint. Thus, removing the low steamline pressure signal will delay SI actuation until the low pressurizer pressure setpoint is reached. The licensee reanalyzed the larger breaks with the SI actuation on low steamline pressure removed. The results showed that the minimum departure from nucleate boiling ratio (DNBR) remains above the regulatory DNBR limit of 1.3, with sufficient margin to conclude that the acceptance criteria for steamline break transient continues to be met with the removal of SI actuation on low steamline pressure.

The licensee evaluated the mass and energy release analysis for a steamline break inside containment to demonstrate that the condition inside containment does not exceed the existing environmental qualification envelope during a steamline break. In this case, regardless of the break size, the SI will actuate on high containment pressure prior to reaching the setpoint for SI actuation on low steamline pressure. Therefore, the removal of the low steamline pressure SI actuation signal does not have any effect on the steamline break mass and energy release as previously reported in the FSAR and UFSAR.

The worst-case scenario for loss of alternating current (AC) power transient results in no primary or secondary depressurization and therefore no SI actuation. However, in the less limiting case, there is the possibility of primary and secondary depressurization due to excessive auxiliary feedwater delivered to the steam generators, compounded by extraction steam loads and the possibility of open steamline drains. The licensee has changed the plant's emergency procedures to include throttling the auxiliary feedwater in the event that 6.9-kV power is unavailable. This prompt operator action will reduce reactor cooling system overcooling and thereby avoid an unnecessary SI actuation. If the throttling action does not occur and overcooling follows, SI actuation on low pressurizer pressure is still available if needed.

The feedwater line break is analyzed to demonstrate long-term cooling and the analysis is required to postulate the break only at the terminal ends of the feedwater piping. For a feedwater line break at the main feedwater pumps, the check valve will prevent depressurization of the steam generator. For a feedwater line break at the steam generator, SI actuation occurs on high containment pressure. Therefore, the elimination of the SI actuation on low steamline pressure does not adversely impact the feedwater line break transient as reported in the UFSAR.

In summary, the licensee's reanalysis or evaluation led to the conclusion that only those transients which involve a secondary system depressurization have the potential to be affected by the elimination of the SI signal on low steamline pressure. For all other transients in which SI actuation occurs, the initiating signal is low pressurizer pressure or high containment pressure, and are thus not affected by the proposed change. Thus all previous acceptance criteria will continue to be met. The staff has reviewed the licensee's submittals and agrees with the licensee's findings. The staff finds the licensee's proposed changes to the Technical Specifications and Bases, as conveyed in the licensee's January 3, 1997, letter acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restrict area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 4345 dated January 29, 1997). The licensee's March 20, 1997, submittal only provides supplemental information and does not change the original amendment request. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

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