APRIL 29, 1997

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

Distribution
Docket File ACRS T-2 E26
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PDII-2 RF GHill(4)
SVarga CCasto,RII
CGrimes JJohnson,RII

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M98107 AND M98108)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 159 to Facility Operating License NPF-35 and Amendment No. 151 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments are in response to your application dated March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997.

The amendments revise Section 3/4.7.1.6 of the Technical Specifications to require four instead of three steam generator pressure operated relief valves operable. The staff reviewed all your submittals under the "technical specifications" provision of 10 CFR 50.59, not the "unreviewed safety question" provision.

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. 159 to NPF-35
2. Amendment No. 151 to NPF-52
3. Safety Evaluation

cc w/encl: See next page

NRC FILE CENTER COPY

DOCUMENT NAME: G:\CATAWBA\CAT98107.AMD

OFFICE	PDII-2/PM	PDII-2/LA	OGC	SPLB/SC	PDII-2/D
NAME	P.TAM:cn	L.BERRY		G.HUBBARD*	H.BERKOW
DATE	4/28/97	4/ /97	4/28/97	4/25/97	4/ /97
COPY	YES NO	YES	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY

*see previous concurrence

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PDR ADOCK 05000413
P PDR

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

April 29, 1997

Mr. William R. McCollum
Site Vice President
Catawba Nuclear Station
Duke Power Company
4800 Concord Road
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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,

Peter S. Tam

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. 159 to NPF-35
2. Amendment No. 151 to NPF-52
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. 159
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 March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 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. 159, which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the license is also hereby amended to add paragraph 2.C.(24) to Facility Operating License No. NPF-35 as follows:

   (24) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 159, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Additional Conditions.

4. This license amendment is effective as of its date of issuance and shall be implemented as follows. The Technical Specification changes shall be implemented within 30 days of issuance of this amendment. The Additional Conditions shall be implemented as stated in Appendix D to the license.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachments: 1. Technical Specification Changes
2. Pages 10, 11 and 12 of License
3. Appendix D - Additional Conditions

Date of Issuance: April 29, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 159

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

Replace the following pages of the Appendix "A" Technical Specifications and the Operating License with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

	Remove	Insert
Appendix A	3/4 7-9 B 3/4 7-3	3/4 7-9 B 3/4 7-3
License	10 11 12	10 11* 12
	---	Appendix D

* overflow page - no change

3.7.1.5 NOT USED

STEAM GENERATOR POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four steam generator power-operated relief valves (PORVs) and associated remote manual controls, including the safety-related gas supply systems, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

a. With one less than the required steam generator PORVs OPERABLE, restore the inoperable steam generator PORV to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required Residual Heat Removal loop in operation for decay heat removal.

b. With two less than the required steam generator PORVs OPERABLE, restore at least one of the inoperable steam generator PORVs to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required Residual Heat Removal loop in operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each steam generator PORV and associated remote manual controls including the safety-related gas supply systems shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that at least one of the two nitrogen bottles associated with each PORV has a pressure greater than or equal to 2100 psig, and

b. At least once per 18 months and prior to startup following any refueling shutdown by verifying that all steam generator PORVs will operate through one cycle of full travel using remote manual controls and safety-related gas supply.

*When steam generators are being used for decay heat removal.

3/4.7.1.6 STEAM GENERATOR POWER OPERATED RELIEF VALVES

The Surveillance Requirement for the Main Steam power-operated relief valves (PORVs) nitrogen supplies ensures that the PORVs will be available to mitigate the consequences of a steam generator tube rupture accident concurrent with loss of offsite power. This assumes that the PORV on the ruptured steam generator is unavailable, and that at least two are used to cool the Reactor Coolant System inventory to less than the saturation temperature of the ruptured steam generator. Local operation of the steam line PORVs is credited in the event that remote operation is unavailable.

Concurrent with the requirement that a specific number of PORVs be OPERABLE is the requirement that the associated PORV block valves upstream be open or OPERABLE. Should an associated PORV block valve be closed and inoperable, the PORV downstream of that block valve should also be considered inoperable and the applicable ACTION statement shall be entered until such time that the block valve is opened or returned to OPERABLE status.

Additionally, if a PORV is inoperable and open, then the requirements of Technical Specification 3.6.3, Containment Isolation Valves, would apply in addition to Technical Specification 3.7.1.6.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

(22) Progress of Offsite Emergency Preparedness (Section 13.3, SER, SSER #1, SSER #2, SSER #3, SSER #4)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(23) Emergency Preparedness Issues (ASLB PID, 9/18/84)

By June 4, 1985, Duke Power Company shall have submitted for staff review and received staff approval on the following items:

1. The Public Information Brochure shall state that high levels of radiation are harmful to health and may be life threatening. Such statements shall be contained within that portion of the brochure that deals with actions to be taken in the event of an emergency.

2. The warning signs and decals shall specify the types of emergencies they cover including nuclear.

3. The warning signs and decals shall notify transients as to where they can obtain local emergency information, as provided in NUREG-0654 Evaluation Criterion II.G.2.

4. The emergency plans shall reflect the kinds of locations within the plume exposure EPZ wherein the warning signs and decals and emergency response information will be placed and the procedures employed to assure that sufficient numbers are being distributed to effectively reach transients, and that the plans are implemented.

5. Comprehensive plans shall provide for early notification to Carowinds of a radiological emergency at Catawba and for evacuation of Carowinds. The plans shall describe the responsibilities of the emergency response organizations of Mecklenburg and York Counties and provide for the coordination of their efforts among themselves and with Carowinds' officials. The plans shall provide for immediate notification of patrons and staff of Carowinds at the time of the precautionary closing of the park, of the cause of the emergency. The means to implement the plans shall be made available.

(24) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 159 , are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendices A, E and J to 10 CFR Part 50. These include (a) partial exemption from General Design Criterion 1 of Appendix A, with respect to the upgrade to safety-related of the pressurizer power operated relief valves (PORVs) and steam generator PORVs until first refueling (Section 5.4.4 of SER and SSER 2, and Section 15.4.4 of SSERs 3 and 4), (b) exemption from the requirements of Appendix E, IV.F, insofar as they may require the active participation of all Crisis Management Center personnel for the Catawba Station emergency preparedness exercises (Section 13.3 of SSER 4), (c) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of the SER, and SSERs 3 and 4), (d) exemption from the requirement of paragraph III.A.(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER 3), and (e) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER 3). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

2.E. Duke Power Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and CFR 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Catawba Nuclear Station Physical Security Plan," with revisions submitted through October 6, 1987; "Catawba Nuclear Station Training and Qualification Plan," with revisions submitted through August 27, 1986; and Catawba Nuclear Station Safeguards Contingency Plan," with revisions submitted through January 8, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Reporting to the Commission

Duke Power Company shall report any violations of the requirements contained in Section 2, Items C.(1), C.(3) through C.(23) of this license. Initial notification shall be made within twenty-four

(24) hours in accordance with the provisions of 10 CFR 50.72 with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c) and (e).

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on December 6, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:
Edson G. Case /for/

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachment 1
2. Appendix A - Technical Specifications
3. Appendix B - Environmental Protection Plan
4. Appendix C - Antitrust Conditions
5. Appendix D - Additional Conditions

Date of Issuance: January 17, 1985

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERTING LICENSE NO. NPF-35

Duke Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
159	This amendment requires the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, and evaluated in the staff's Safety Evaluation dated April 29, 1997.	Next update of the UFSAR
159	This amendment requires the licensee to use administrative controls, as described in the licensee's letter of March 7, 1997, and evaluated in the staff's safety evaluation dated April 29, 1997, to restrict the dose-equivalent iodine levels to 0.46 microCurie per gram (in lieu of the limit in TS Section 3.4.8.a), and to 26 microCurie per gram (in lieu of the limit of TS Figure 3.4-1), until this license condition is removed by a future amendment.	Immediately upon issuance of the amendment

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. 151
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 March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 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. 151, which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the license is also hereby amended to add paragraph 2.C.(24) to Facility Operating License No. NPF-52 as follows:

   (24) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 151, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Additional Conditions.

4. This license amendment is effective as of its date of issuance and shall be implemented as follows. The Technical Specification changes shall be implemented within 30 days of issuance of this amendment. The Additional Conditions shall be implemented as stated in Appendix D to the license.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachments: 1. Technical Specifications Changes
2. Pages 6 and 7 of the License
3. Appendix D

Date of Issuance: April 29, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications and the Operating License with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

	Remove	Insert
Appendix A	3/4 7-10 B 3/4 7-3	3/4 7-10 B 3/4 7-3
License	6 7	6 7
	---	Appendix D

PLANT SYSTEMS

3.7.1.5 NOT USED

STEAM GENERATOR POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four steam generator power-operated relief valves (PORVs) and associated remote manual controls, including the safety-related gas supply systems, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.*

ACTION:

a. With one less than the required steam generator PORVs OPERABLE, restore the inoperable steam generator PORV to OPERABLE status within 7 days; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required Residual Heat Removal loop in operation for decay heat removal.

b. With two less than the required steam generator PORVs OPERABLE, restore at least one of the inoperable steam generator PORVs to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours and place the required Residual Heat Removal loop in operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each steam generator PORV and associated remote manual controls including the safety-related gas supply systems shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that at least one of the two nitrogen bottles associated with each PORV has a pressure greater than or equal to 2100 psig, and

b. At least once per 18 months and prior to startup following any refueling shutdown by verifying that all steam generator PORVs will operate through one cycle of full travel using remote manual controls and safety-related gas supply.

*When steam generators are being used for decay heat removal.

3/4.7.1.6 STEAM GENERATOR POWER OPERATED RELIEF VALVES

The Surveillance Requirement for the Main Steam power-operated relief valves (PORVs) nitrogen supplies ensures that the PORVs will be available to mitigate the consequences of a steam generator tube rupture accident concurrent with loss of offsite power. This assumes that the PORV on the ruptured steam generator is unavailable, and that at least two are used to cool the Reactor Coolant System inventory to less than the saturation temperature of the ruptured steam generator. Local operation of the steam line PORVs is credited in the event that remote operation is unavailable.

Concurrent with the requirement that a specific number of PORVs be OPERABLE is the requirement that the associated PORV block valves upstream be open or OPERABLE. Should an associated PORV block valve be closed and inoperable, the PORV downstream of that block valve should also be considered inoperable and the applicable ACTION statement shall be entered until such time that the block valve is opened or returned to OPERABLE status.

Additionally, if a PORV is inoperable and open, then the requirements of Technical Specification 3.6.3, Containment Isolation Valves, would apply in addition to Technical Specification 3.7.1.6.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

(12) Generic Letter 83-28 (Section 15.6, SSER #4, SSER #5)

Duke Power Company shall submit responses to and implement the guidance of Generic Letter 83-28 on a schedule which is consistent with that given in its November 2 and December 31, 1984, letters.

(13) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 151, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below, and pursuant to evaluations contained in the referenced SER and SSERs. These include (a) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of SSER #5), (b) exemption from the requirement of paragraph III.A.1(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #5), and (c) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and certain special circumstances, as discussed in Section 6.2.6 of SSER #5, are present. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission. In addition, two exemptions were previously granted pursuant to 10 CFR 50.12. A partial exemption from those portions of General Design Criterion 4 of Appendix A to 10 CFR 50 which require protection of structures, systems and components important to safety against dynamic effects associated with postulated reactor coolant system pipe breaks was granted on April 23, 1985, for a period ending with the completion of the second refueling outage for Catawba Unit 2 or the adoption of the proposed rulemaking for modification of GDC-4 whichever occurs first. Effective May 12, 1986, GDC-4 has been modified to exclude from the design basis the protection of structures, systems and components against the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in PWRs when analyses demonstrate the probability of rupture of such piping to be extremely low under design basis conditions (51 FR 12502 April 11, 1986). As a result of this final rule and Duke Power Company's demonstration in accordance with the rule, the previously granted specific partial exemption will no longer be required, on the rule's effective date, and terminate by its own terms. Furthermore, an exemption from the requirements of Appendix E, IV.F, insofar as they may require the active participation of all Crisis Management Center personnel for the Catawba Station emergency preparedness exercises (Section 13.3 of SSER #4), was granted on January 17, 1985, by the issuance of Facility Operating License No. NPF-35 for Catawba Nuclear Station, Unit 1.

2.E. Duke Power Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and CFR 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Catawba Nuclear Station Physical Security Plan," with revisions submitted through October 6, 1987; "Catawba Nuclear Station Training and Qualification Plan," with revisions submitted through August 27, 1986; and Catawba Nuclear Station Safeguards Contingency Plan," with revisions submitted through January 8, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Reporting to the Commission

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Duke Power Company shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty-four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c), and (e).

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on February 24, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachment 1
2. Appendix A - Technical Specifications
3. Appendix B - Environmental Protection Plan
4. Appendix C - Antitrust Conditions
5. Appendix D - Additional Conditions

Date of Issuance: May 15, 1986

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERTING LICENSE NO. NPF-52

Duke Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
151	This amendment requires the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, and evaluated in the staff's Safety Evaluation dated April 29, 1997.	Next update of the UFSAR
151	This amendment requires the licensee to use administrative controls, as described in the licensee's letter of March 7, 1997, and evaluated in the staff's safety evaluation dated April 29, 1997, to restrict the dose-equivalent iodine levels to 0.46 microCurie per gram (in lieu of the limit in TS Section 3.4.8.a), and to 26 microCurie per gram (in lieu of the limit of TS Figure 3.4-1), until this license condition is removed by a future amendment.	Immediately upon issuance of the amendment

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 159 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 151 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 March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, Duke Power Company (DPC, the licensee) proposed changes to the plant Technical Specifications (TSs), the Updated Final Safety Analysis Report (UFSAR), and associated Selected Licensee Commitments (SLC) document for the Catawba Nuclear Station, Units 1 and 2. Specifically, the licensee proposed to revise TS 3/4.7.1.6, "Steam Generator Power Operated Relief Valves," SLC 16.10-1, "Steam Vent to Atmosphere," and UFSAR Chapter 15.6.3, "Steam Generator Tube Rupture," to require four instead of three steam generator power operated relief valves (PORVs) and their gas supply systems to be operable. The need for the proposed change was identified by the licensee during a self-initiated review of dose methodologies, assumptions, inputs, and results in Chapter 15 of the UFSAR.

The April 2, 10, 16, 22, and 28, 1997, letters provided clarifying information that did not change the scope of the March 7, 1997, application and the initial proposed no significant hazards consideration determination.

Catawba is a four-loop Westinghouse plant with a PORV on each of its four steam generators. During its review, the licensee discovered that a single failure of a safety-related power supply would cause the loss of power to two of these PORVs and limit the capability to cooldown and depressurize the plant in accordance with the UFSAR Chapter 15.6.3 analysis of steam generator tube rupture (SGTR) accidents. The licensee also determined that even without this new single failure discovery, the current TS for steam generator PORVs was inadequate in that it only requires three PORVs to be operable. Therefore, if one of the three operable PORVs is on the steam generator with the SGTR, a single failure of the safety-related power supply to the remaining two operable PORVs would leave only one available PORV. In the current SGTR analysis, the licensee assumes that two PORVs, other than the PORV on the affected steam generator, can be remotely operated from the control room. To correct this deficiency, the licensee has proposed to revise TS 3.7.1.6 to require four operable PORVs in lieu of three. To be consistent with the change to TS 3.7.1.6, the licensee also proposed to revise SLC 16.10-1 to require four PORV steam generator safety-related gas supply systems to be operable in lieu of three. Each solenoid-actuated, air-operated PORV has a

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back-up safety-related gas supply system that uses two nitrogen bottles for PORV operation in the event of a loss of the normal air supply from the instrument air system.

To resolve the concern associated with a single power supply failure causing two PORVs to be inoperable from the control room, the licensee has proposed changes to the licensing basis as described in the UFSAR and the Bases section for TS 3.7.1.6 to allow credit for operator action to open the second PORV locally following an SGTR with a coincident failure of a safety-related power supply that affects two PORVs. In support of these changes, the licensee has performed dose analyses and steam generator overfill analyses to justify the time delay for local operator action in lieu of remote operator action in the control room to open the PORVs. The licensee has also provided information from a human factors standpoint as justification that the operators can adequately perform the required actions within the time frame assumed in the new analyses.

2.0 DISCUSSION AND EVALUATION

2.1 Proposed Change to the Plant Technical Specifications And Bases

The proposed change to TS 3.7.1.6 requiring four PORVs and their associated gas supply systems to be operable is necessary to ensure at least two PORVs are available following a single failure as assumed in the current and revised SGTR analysis. With only one PORV available following an SGTR, steam generator overfill could occur in the affected steam generator because of the extended time for primary plant cooldown and depressurization. An acceptance criterion used by the licensee for the SGTR analysis is that steam generator overfill does not occur. The proposed change results in an increase in the number of PORVs required to be operable, thereby reflecting an increase in the level of safety. The proposed change also revises the specification to coincide with the actual plant design and accident analyses. The proposed change is, therefore, acceptable.

Each PORV has a safety-related nitrogen backup system (two bottles), referred to as the gas supply systems, which is relied upon for operation of the PORVs when the instrument air system is not available. Currently, SLC 16.10-1 specifies that three steam generator PORV safety-related gas supply systems shall be operable with both nitrogen bottles per PORV pressurized to greater than or equal to 2100 pounds per square inch (psi). The licensee's proposed change to SLC 16.10-1 would specify four gas supply systems to be operable in lieu of three. This change is consistent with the changes to TS 3.7.1.6 requiring four PORVs and their associated gas supply systems to be operable. The proposed change is acceptable on the same basis as previously discussed for the proposed PORV TS change.

The proposed changes to the UFSAR and the Bases section for TS 3.7.1.6 resulted from the licensee's discovery of potential single failures associated with the control power system for the PORVs, which result in a loss of control power to two PORVs. These control power failures would render two PORVs incapable of being operated from the control room. However, local operation of the PORVs (i.e., with handwheels) is not affected by these failures. The licensee, therefore, proposed changes to reflect local manual operation of a

PORV on an intact steam generator in conjunction with remote operation of the PORV on the remaining intact steam generator that was not affected by the control power failure. This action would only have to be taken for an SGTR with a loss of offsite power coupled with specific postulated control power system failures that could affect two PORVs. The licensee performed new analyses of the SGTR dose assessment and steam generator overfill potential, and evaluated the operator's capability to perform the required actions within the time constraints assumed in the new analyses. The staff's review is set forth below.

2.2 Revised Licensing Bases as Documented in the UFSAR

2.2.1 Steam Generator Overfill Analysis

The licensee performed detailed SGTR overfill analyses using the RETRAN-02 computer code. The RETRAN-02 computer code was approved for SGTR analyses for Catawba Units 1 and 2 in a safety evaluation report (letter, R. Martin to M. Tuckman, dated December 28, 1995), which approved Topical Report DPC-NE-3002, Revision 1, "FSAR Chapter 15 System Transient Analysis Methodology." Approval of the RETRAN-02 code was limited to SGTR scenarios where subcooling in the primary system is maintained and, therefore, two-phase flow is not encountered. The licensee's analyses were performed within the approved limits and conditions defined in the topical report and, therefore, the use of the RETRAN-02 computer code for these analyses was consistent with this prior approval.

In the new analyses, the licensee assumed that all four SG PORVs were initially operable and available for remote operation from the control room. A single failure was assumed to render PORVs on two intact SGs incapable of being remotely operated from the control room. This failure left only one PORV on an intact SG capable of being remotely controlled from the control room. Therefore, local manual operation of another PORV on an intact SG was assumed with associated time delays. The licensee has conducted simulator tests that showed that operators can begin local manual operation of an SG PORV and otherwise respond as assumed in its analyses. In addition, the emergency operating procedure used for the SGTR accident currently includes directions to dispatch an operator to manually operate SG PORVs to effect plant cooldown. Unit-specific assumptions were used in the analyses to maximize primary-to-secondary leakage through the ruptured tube and conservatively modeled overfill. This was necessary because of the differences in design and operation of the SGs for the units (Unit 1 has Babcock & Wilcox International (BWI) SGs while Unit 2 has Westinghouse (W) Model D5 SGs). The following is a brief discussion of these assumptions.

Unit 1

- The Catawba Unit 1 BWI SGs have a constant program level for a power range of 25% to 100%. Therefore, since initial power level would have no effect on SG secondary mass, a +2% overpower factor was conservatively applied in the Unit 1 analysis.

- An initially high reactor coolant system (RCS) pressure was assumed for Unit 1. An early manual reactor trip is assumed in the Unit 1 analysis and, therefore, the high RCS pressure assumption will not affect the reactor trip. However, this assumption will provide for conservatively higher primary-to-secondary leakage as a result of the higher primary system pressure.

- A high pressurizer water level was assumed to maximize primary-to-secondary leakage.

- A high SG water level was assumed at the time of the reactor trip. Since the program level for the Catawba Unit 1 BWI SGs is constant from 25% to 100% power, that level (65% narrow range), was assumed with instrument uncertainty added in the positive direction to maximize SG level. This assumption results in less margin to overfill and is, therefore, conservative.

- Main steam safety valve (MSSV) and SG PORV setpoints were assumed low with -3% drift on the MSSV setpoints and -75 psi reduction in the SG PORV setpoint. This increases primary-to-secondary differential pressure and, thereby increases primary-to-secondary leakage.

- Auxiliary feedwater (AFW) start time was assumed conservatively fast while flow rates were assumed conservatively high. Additionally, the worst flow imbalance with the highest flow rate to the ruptured SG was assumed. AFW was assumed to be throttled at an SG narrow range level of 39% (plus instrument uncertainty in the positive direction) for the Unit 1 BWI SGs.

- Safety Injection (SI) was assumed to be manually actuated at the time of the trip for Unit 1. Manual actuation was assumed to occur sooner than it would have taken the RCS to reach the automatic low pressure SI actuation signal and is therefore conservative. SI flow rates were assumed conservatively high to maximize primary-to-secondary leakage.

- The break location was assumed at the top of the tube sheet on the cold side of the SGs. This maximizes primary-to-secondary leakage.

- Decay heat for 102% of full power with a 2 sigma uncertainty was conservatively assumed.

Unit 2

- A full power level without an overpower factor was assumed to maximize SG secondary mass. Application of an overpower factor was not applied because the secondary mass in the Unit 2 W D5 SGs decreases with increasing power level.

- Low initial RCS pressure was assumed to minimize the time to reactor trip and allow for a turbine runback. This combination results in increasing SG secondary mass for the Unit 2 W Model D5 SGs prior to the reactor trip.

- A high pressurizer water level was assumed to maximize primary-to-secondary leakage.

- A high SG water level was assumed at the time of the reactor trip. A turbine runback would result in increasing the secondary side mass of the W Model D5 SGs. A turbine runback would result in a power level of approximately 67% at the time of the reactor trip. A more conservative SG level reflective of 55% power was assumed in the analysis. In addition, instrument uncertainty was accounted for in the positive direction to maximize SG level. This assumption results in less margin to overfill and is, therefore, conservative.

- MSSV and SG PORV setpoints were assumed low with -3% drift on the MSSV setpoints and -75 psi reduction in the SG PORV setpoint. This increases primary-to-secondary differential pressure and, thereby increases primary-to-secondary leakage.

- AFW start time was assumed conservatively fast while flow rates were assumed conservatively high. Additionally, the worst flow imbalance with the highest flow rate to the ruptured SG was assumed. AFW was assumed to be throttled at a SG narrow range level of 62% (plus instrument uncertainty in the positive direction) for the Unit 2 W Model D5 SGs.

- SI was assumed to actuate on the low RCS pressure SI signal for Unit 2. SI start time was assumed conservatively fast while flow rates were assumed conservatively high, both in order to maximize primary-to-secondary leakage.

- The break location was assumed at the top of the tube sheet on the cold side of the SGs. This maximizes primary-to-secondary leakage.

- Decay heat for 102% of full power with a 2 sigma uncertainty was conservatively assumed.

Staff review of the licensee's submittals finds the assumptions used in the analyses conservative with respect to SG overfill. The review has also confirmed that the licensee's use of the RETRAN-2 computer code to perform these analyses consistent with the prior approval of the code for Catawba. Therefore, the staff concludes that the licensee's analyses conservatively model the SGTR for potential SG overfill. The licensee's analyses have shown that margin to overfill exists, hence, overfill does not occur. Based on the above discussion, the staff finds the licensee's analyses for potential SG overfill acceptable. In addition, the staff finds the proposed changes to TS 3/4.7.1.6 and SLC Section 16.10-1 (to increase the number of SG PORV required to be operable from three to four) consistent with the new analyses, and are required to ensure that SG overfill does not occur as conservatively analyzed in the new analyses.

2.2.2 Operator Actions

Standard Review Plan Section 5.4.7, "Residual Heat Removal (RHR) System," states: "In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified." Standard Review Plan

Chapter 15, "Accident Analysis," indicates that operator actions for the SGTR accident were anticipated as part of the design-basis analyses.

In those instances in which licensees consider temporary or permanent changes to the facility that credit operator actions for previously automatic system or component actuations, the staff has evaluated such changes using guidance in Generic Letter 91-18, Section 6.7, "Use of Manual Action in Place of Automatic Action," and ANSI/ANS Standard 58.8, "Time Response Design Criteria for Safety Related Operator Actions," 1984. The staff also used plant-specific review criteria (Letter, C.E. Rossi to A.E. Ladieu, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill" WCAP-10698, March 30, 1987) for assessing operator action times in the event of an SGTR.

Generic Letter 91-18 states: "The consideration of manual action in...areas also must include the ability and timing in getting to the area, training of personnel to accomplish the task, and occupational hazards to be incurred such as radiation, temperature, chemical, sound, or visibility hazards." ANSI-58.8 supplies estimates of reasonable response times for operator actions, but does allow licensees to use time intervals derived from independent sources, provided they are based on task analyses with consideration given to human performance. The staff evaluated the licensee's task-analysis-related responses as follows.

(1) Specific operator actions required - The licensee noted the operator actions required for local SG PORV operation of SGs in Enclosure 3 of its submittal. These actions included using emergency equipment and communicating with the control room.

(2) Potentially harsh or inhospitable environmental conditions expected - The licensee stated that the environmental conditions in the area in which the PORV-related manipulations will occur are not expected to be inhospitable or harsh.

(3) General discussion of the ingress/egress paths taken by the operators to perform functions - In Enclosure 3 of its submittal, the licensee discussed the preferred routes to take for performing the local manual actions.

(4) Procedural guidance for required actions - The licensee submitted plant procedure EP/1/A/5000/E-3, which documented the required local manual actions.

(5) Specific operator training necessary to carry out actions including any operator qualifications required to carry out actions - The licensee stated that operator training included detailed instruction on how to manually engage the PORV and operate it locally. The licensee also stated that nonlicensed operator task N-0092, element #4 (walk through local operation of SG PORV) is part of the initial job training of each nonlicensed operator.

(6) Any additional support personnel and equipment required by the operator to carry out actions - The licensee stated that operators need no additional support personnel to perform the local manual tasks

and listed required equipment. The licensee also noted that the task requires two operators and both are assumed present in the control room.

(7) Description of information required by the control room staff to determine such operator action is required, including qualified instrumentation[1] used to diagnose the situation and to verify that the required action has been successfully taken - The licensee stated that successful completion of the local manual action will appear through the safety-related steam pressure, reactor coolant temperature instrumentation, and position indication lights of the SG PORV.

(8) Ability to recover from plausible errors in performance of manual actions, and the expected time required to make such a recovery - The licensee acknowledged that the most plausible error would be for operators to go to the wrong PORV. However, the licensee added that this is unlikely, not only because two operators perform the task, but also because of communication established with the control room.

The staff finds the previously discussed information acceptable because it is consistent with Standard Review Plan guidance, ANSI-58.8, and Generic Letter (GL) 91-18 and, as discussed further below, shows that the limited operator action will not affect the ability of the system to perform its function.

Required operator actions and time associated with the local manual operation as provided by the licensee are as follows:

Actions	Expected Time (in minutes)	Demonstrated Time (in minutes)
Travel to the PORV	10	8
Engage in clevis (pin)	2	1
Fully open the PORV	6	4
Total	18*	13

* Credit in the licensee's analysis is taken for completion of the local manual action at 41 minutes.

The licensee's analysis has sufficiently considered the important contribution of operator errors of omission or commission (e.g., going to the wrong PORV) which could have delayed proper response and potentially increase the consequences of failure to manually operate one of the SG PORVs affected by a

1 In accordance with RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants Assess Plant and Environs Conditions During and Following an Accident," Revision 3, 1983, qualification of the instrumentation relied upon by the operators may be an important review issue. RG 1.97, defines Type A variables as: "those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their functions for design-basis accident events."

postulated single failure. The expected time available for operator action in this situation was approximately 18 minutes to make at least one pressurizer PORV available by opening its associated block valve. The demonstrated time to make one pressurizer PORV available was 13 minutes. The licensee's determination that 18 minutes was sufficient for successfully completing the local manual action, along with an assumed time of 41 minutes in its analysis, is consistent with ANSI-58.8 guidance, which suggests allowing at least 30 minutes for operator actions outside the control room. The staff finds, therefore, that the times shown in the above table are acceptable because the demonstrated times were bounded by the assumed times, the total time is consistent with ANSI-58.5, and the licensee documented previously discussed task-analysis-related information as the basis for its time determinations.

Although GL 91-18 does not explicitly provide that licensees are to analyze the consequences of operator performance errors and the likelihood of recovering from such errors, the staff expects licensees to consider the possibility of operator errors and determine if sufficient time exists to recover from such errors. This staff position is based, in part, on ANSI-58.8, which states that:

> Nuclear-safety-related operator actions or sequences of actions may be performed by an operator only where a single operator error of one manipulation does not result in exceeding the design requirements for design-basis events.

It does appear that the licensee's evaluation considered the possibility of performance errors or the likelihood of recovering from such errors given the time frame allotted for accomplishing the manual required actions. Given the duration of time available for operator action, it appears likely that recovery from an error in performance could be achieved without exceeding the required 18 minutes to make at least one PORV available by opening its associated block valve.

The staff's safety evaluation (letter, R. Martin to M. Tuckman, dated May 14, 1991) concluded that the licensee's demonstrated times for Catawba operators were satisfactory. The licensee's letter of April 2, 1997, updated its responses relevant to an SGTR and operator action times. The operator action times relevant to the SGTR analysis are the subject of the present review.

Criterion 1. Provide simulator and emergency operating procedure training related to a potential SGTR.

The licensee documented by letter dated April 2, 1997, that the subject training relevant to an SGTR is provided. The staff finds that the licensee has satisfied Criterion 1.

Criterion 2. Provide plant-specific operator response times.

The licensee provided plant-specific operator response times as shown in the following table:

Operator Action	Assumed Time (in minutes after reactor trip unless otherwise noted)	Demonstrated Time (in minutes)
Throttel AFW	5.9 (Unit 1) 11.5 (Unit 2)	11.5
Identify and isolate the required SG	13	14*
Begin Cooldown	23	21
Reach the second PORV	33	31
Second PORV begins to open	35	33
Second PORV fully open	41	37
Begin depressurizing	3 after cooldown	2
Terminate safety injection	3 after depressurization	3

* Identified ruptured SG in 12 minutes.

The demonstrated times in this table were bounded by the assumed times with one exception. This exception concerns the assumed time (13 minutes) and the demonstrated time (14 minutes) for identifying and isolating the ruptured SG. The staff considered the difference between these two times to be acceptable because (1) the licensee stated that the ruptured SG could be identified in 12 minutes, and (2) the licensee's evaluation indicated margin to overfill would be maintained. The staff finds that the licensee has satisfied Criterion 2.

Criterion 3. Utilizing typical control room staff as participants in demonstration runs, show that the operator action times assumed in the SGTR analysis are realistic.

The licensee's April 16, 1997, submittal indicated that about three trial runs were completed. The licensee noted that the SGTR accident is one of the simulator events on which operators are regularly tested. Further, the licensee stated that two operators would be dispatched to perform the required local actions. This would allow one operator to perform the local manual action and one to maintain communications with the control room. The licensee responded that (1) the time from initiation of a tube rupture to the point of safety injection (SI) termination ranges from 55 to 60 minutes, (2) time validations have shown SI termination times from 34 to 48 minutes, with the average at about 45 minutes and 48 minutes for 1994 and recent scenarios, respectively. On the basis that the times achieved during the time validations were bounded by the times expected for SI termination, the staff finds that the licensee has satisfied Criterion 3.

Criterion 4. Complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention, using design-basis assumptions regarding available equipment.

As noted under Criterion 2, the licensee's demonstrated times were bounded by assumed times, and the staff finds that the licensee has satisfied Criterion 4.

Criterion 5. If the emergency operating procedures (EOPs) specify SG sampling as a means of identifying the SG with the ruptured tube, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

The licensee's letter of April 16, 1997, indicated that the EOPs for Catawba Units 1 and 2 do not specify SG sampling as a means of identifying a ruptured SG. Instead, steamline monitors and SG level are means of identifying a ruptured SG. On the Basis of this information, the staff finds that Criterion 5 is satisfied.

The staff finds the licensee's proposed manual operator actions and associated action times acceptable as reviewed in detail above.

2.2.3 Radiological Consequences of a Postulated Steam Generator Tube Rupture Accident

The licensee assessed the consequences of an SGTR coincident with a loss of offsite power occurring concurrent with a reactor trip. The single failure postulated for the SGTR was assumed to occur within a channel of the Vital Instrumentation and Control Systems resulting in a loss of control power to two SG PORVs. The licensee plans a full-fledged thermal hydraulic assessment of this postulated accident later this year; results of such an assessment would be factored into the reanalysis of SGTR radiological consequences. For the interim, in the absence of such detailed thermo hydraulic assessment, the licensee self-imposed (see licensee's letter of March 7, 1997) administrative limits on dose equivalent ^{131}I levels which are more restrictive than the current TS limits. The staff thus calculated radiological doses for two cases of this accident. The first case assumed that the SGTR event occurred following a pre-existing spike with the primary coolant activity level of dose equivalent ^{131}I equal to 26 $\mu Ci/g$ (in lieu of the 60 $\mu Ci/g$ in the TS). The second case assumed that the SGTR accident initiated a spike (i.e., a 500-fold increase of iodine release from the fuel) and that the primary coolant activity level of dose equivalent ^{131}I at the time of initiation was 0.46 $\mu Ci/g$ (in lieu of the 1.0 $\mu Ci/g$ in the TS).

The licensee assumed that primary-to-secondary leakage occurred at the TS limits of 150 gpd/SG with total leakage limited to 0.4 gpm. The licensee did not perform a thermal hydraulic analysis for the scenario identified above. Instead the licensee took the results of a RETRAN-02 analysis of an SG overfill case and projected it on a timeline associated with the mitigation of radiological consequences.

For the determination of flash fraction, the primary side conditions were taken from the previously referenced RETRAN-02 analysis. The licensee assumed lower bounds on secondary side pressure, 920 psia before trip, 850 psia for the first 36 minutes after trip and 970 psi afterwards. This information was based upon an analysis of an SG overfill occurring following an SGTR at

Unit 2. The licensee indicated that these pressure values are lower than the calculated values of transient secondary side pressures of either Unit 1 or Unit 2.

The licensee assumed that bypass occurred for the first 5 minutes following reactor trip. A bypass fraction of 12% was assumed based on a review of the analysis of dose inputs, WCAP-13132 and previous analyses of the radiological consequences of SGTR accidents. The licensee indicated that this assumption was made as a conservatism to provide analytical margin in the absence of a precise analysis of the thermal hydraulic input of the dose analysis. The licensee did not anticipate any bypass flow when the detailed thermal hydraulics analysis is performed. In its interim dose assessment, the licensee lumped together the bypass fraction and the flashing fraction to give a combined value for the flash and bypass releases.

Additional information concerning the licensee's assumptions for the SGTR are contained in a table to the licensee's April 2, 1997, submittal. Revised information on break flow, flashing fraction, steaming rates, all as a function of time, were provided in the April 22, 1997, submittal.

The staff performed a confirmatory analysis to demonstrate that, in the event of an SGTR accident, the Catawba Nuclear Plant will not generate releases which would lead to doses exceeding the guidelines in SRP 15.6.3. The assumptions, which the staff used in its assessment, are presented in the attached table; the doses calculated by the staff are presented in the table after that. The staff's calculations showed that for the pre-existing spike case, doses were less than Part 100 at offsite locations and within the guideline limits of General Design Criterion 19 for the control room operator. For the accident-initiated spike case, the staff concluded that the postulated doses would be within a small fraction of Part 100 as detailed in SRP 15.6.3.

As previously stated, the staff's dose calculation was based on the licensee's current administrative limits on dose-equivalent iodine levels (0.46 and 26 microCurie per gram) as indicated in the licensee's March 7 and April 2, 1997, letters, and not on the current TS limits imposed in Section 3.4.8.a and Figure 3.4-1. The staff will, thus, impose a license condition to be located in a new Appendix D to the operating license to affirm the licensee's current administrative limits in lieu of the TS limits. The adequacy of the TS limits under 10 CFR 50.36 is an unreviewed issue pending a determination of their validity, or a revision of the limits based on the licensee's future thermal hydraulic assessment results to be submitted later this year. Until resolution of this issue by a future amendment, the licensee's administrative limits, more restrictive than the TS limits, shall govern.

3.0 SUMMARY

As set forth in the above sections, the staff found the licensee's proposed changes to the TS, TS Bases, the UFSAR and associated SLC document acceptable. The staff's findings were based on reviewing the licensee's analyses addressing potential steam generator overfill, needed local manual operation of one of the PORVs on an intact SG, and a radiological dose analysis using the licensee's current restrictive administrative limits on dose-equivalent ^{131}I limits. The staff will impose two new license conditions, to be located in a new Appendix D of the operating license, to require that the licensee update the UFSAR as committed, and to require that the plant be operated with the administrative dose-equivalent ^{131}I limits. By a letter dated April 28, 1997, the licensee indicated it has no objection to these license conditions.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 11931 dated March 13, 1997). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The staff concludes, 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.

Attachment: Table

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Peter S. Tam

Date: April 29, 1997

INPUT PARAMETERS FOR EVALUATION OF SGTR AT CATAWBA

1. Primary and secondary coolants:

Primary Coolant Volume (ft^3)	11,259
Primary Coolant Temperature (°F)	585.1
Secondary Coolant Steam Volume (ft^3)	3,742
Primary Coolant Pressure (psia)	2250
Primary Coolant Mass (lbs)	536,705
Pressurizer Volume (ft^3)	996.34
Pressurizer Temperature (°F)	652.5
Pressurizer Pressure (psia)	2250
Secondary Coolant Mass/SG (lbs)	116,500
Secondary Steam Temperature (°F)	652.5

2. Limits for dose-equivalent ^{131}I in the primary and secondary coolants:

Maximum Instantaneous in primary coolant (μCi/g)	26
48 Hour DE in primary coolant (μCi/g)	0.46
Secondary Coolant (μCi/g)	0.1

3. TS value for the primary to secondary leak rate:

Any steam generator (gpd)	150
Total all SGs (gpm)	0.4

4. Iodine Partition Factor

Faulted SG	0.01
Intact SG	0.01
Condenser	0.15

5. Steam Released to the environment: Refer to April 22, 1997, letter from licensee

6. Letdown Flow Rate (gpm): 75

7. Atmospheric Dispersion Factors:

Exclusion Boundary (EAB, 0-2 hours)	4.78×10^{-4}
Low Population Zone (EPZ, 0-8 hours)	6.85×10^{-5}
Control Room (0-8 hours)	1.0×10^{-2}

8. Control Room:

Emergency Makeup Flow (cfm)	1,800
Makeup Filter efficiency (%)	99
Unfiltered Inleakage (cfm)	10
Recirculation Filter Flow Rate (cfm)	3,600
Recirculation Filter Efficiency (%)	99
Occupancy Factor (0-1 day)	1.0

9. Flashing Fraction — Refer to licensee's April 22, 1997, letter

STEAM GENERATOR TUBE RUPTURE THYROID DOSE ASSESSMENT

Case Involving Pre-existing Spike

	EAB	LPZ	Control Room
Calculated thyroid dose (rem)	63	10	12.7
Regulatory Limits (rem)	300	300	30

Case Involving No Pre-Existing Spike

	EAB	LPZ	Control Room
Calculated thyroid dose (rem)	16	2.8	3.4
Regulatory Guidelines (rem)	30	30	30