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

December 30, 2008

Mr. J. R. Morris
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM COMPLETION TIMES, BYPASS TEST TIMES AND SURVEILLANCE TEST INTERVALS (TAC NOS. MD7718 AND MD7719)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 247 to Renewed Facility Operating License NPF-35 and Amendment No. 240 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 11, 2007, as supplemented by letter dated December 18, 2008.

The amendments revise several Technical Specification sections to allow the bypass test times and Completion Times (CTs) for Limiting Condition for Operation (LCOs) 3.3.1, "Reactor Trip System (RTS) Instrumentation;" 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation;" 3.3.6, "Containment Air Release and Addition Isolation Instrumentation," and 3.3.9, "Boron Dilution Mitigation System [BDMS]."

The proposed license amendment request (LAR) adopts changes as described in Westinghouse Commercial Atomic Power (WCAP) Topical Report WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the Reactor Protection System and Engineered Safety Features Actuation System Test Times and Completion Times," issued October 1998 and approved by U.S. Nuclear Regulatory Commission (NRC) letter dated July 15, 1998. Implementation of the proposed changes is consistent with Technical Specification Task Force (TSTF) Traveler TSTF-418, Revision 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)." The NRC approved TSTF-418, Revision 2, by letter dated April 2, 2003.

In addition, the proposed LAR adopts changes as described in WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," issued March 2003, as approved by NRC letter dated December 20, 2002. Implementation of the proposed changes is consistent with TSTF Traveler TSTF-411, Revision 1, "Surveillance Test Interval Extension for Components of the Reactor Protection System (WCAP-15376)." The NRC approved TSTF-411, Revision 1, by letter dated August 30, 2002. The licensee also requested additional changes not specifically included in the above topical reports. These changes will be evaluated in a future amendment.

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.

If you have any questions, please call me at 301-415-1345.

Sincerely,

John Stang

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 247 to NPF-35
2. Amendment No. 240 to NPF-52
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
Renewed 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) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Carolinas, LLC, acting for itself, and North Carolina Electric Membership Corporation (licensees), dated December 11, 2007, as supplemented December 18, 2008, 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.

Enclosure 1

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed 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. 247, which are attached hereto, are hereby incorporated into this license. Duke Energy Carolinas, LLC, 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 within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. NPF-35
and the Technical Specifications

Date of Issuance: December 22, 2008

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
Renewed 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) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated December 11, 2007, as supplemented December 18, 2008, 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.

Enclosure 2

2. Accordingly, the license is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed 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. 240, which are attached hereto, are hereby incorporated into this license. Duke Energy Carolinas, LLC, 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 within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to License No. NPF-52 and the Technical Specifications

Date of Issuance: December 22, 2008

ATTACHMENT TO LICENSE AMENDMENT NO 247

RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 240

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License Pages	License Pages
NPF-35 page 4	NPF-35 page 4
NPF-52 page 4	NPF-52 page 4
TSs	TSs
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-10	3.3.1-10
3.3.1-11	3.3.1-11
3.3.2-2	3.3.2-2
3.3.2-3	3.3.2-3
3.3.2-4	3.3.2-4
3.3.2-5	3.3.2-5
3.3.2-9	3.3.2-9
3.3.6-2	3.3.6-2
3.3.9-3	3.3.9-3

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 247 which are attached hereto, are hereby incorporated into this renewed operating license Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

---

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplement wherein this renewed license condition is discussed.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 240 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	------------------NOTE------------------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment of other channels. --	
	D.1.1 -------------NOTE-------------- Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ----------------------------------- Perform SR 3.2.4.2.	12 hours from discovery of THERMAL POWER > 75% RTP AND Once per 12 hours thereafter
	AND	
	D.1.2 Place channel in trip.	72 hours
	OR	
	D.2 Be in MODE 3.	78 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ---	
	E.1 Place channel in trip.	72 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	78 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	24 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	24 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1 --------------NOTE-------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. -------------------------------------	
	Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. One channel inoperable.	-------------------NOTE------------------ The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ---	
	L.1 Place channel in trip.	72 hours
	<u>OR</u>	
	L.2 Reduce THERMAL POWER to < P-7.	78 hours
M. One Reactor Coolant Flow - Low (Single Loop) channel inoperable.	------------------NOTE------------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---	
	M.1 Place channel in trip.	6 hours
	<u>OR</u>	
	M.2 Reduce THERMAL POWER to < P-8.	10 hours
N. One Turbine Trip - Stop Valve EH Pressure Low channel inoperable.	------------------NOTE------------------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ---	
	N.1 Place channel in trip.	72 hours
	<u>OR</u>	
	N.2 Reduce THERMAL POWER to < P-9.	76 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One or more Turbine Trip - Turbine Stop Valve Closure channels inoperable.	O.1 Place channel(s) in trip.	72 hours
	OR	
	O.2 Reduce THERMAL POWER to < P-9.	76 hours
P. One train inoperable.	------------------NOTE------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --	
	P.1 Restore train to OPERABLE status.	24 hours
	OR	
	P.2 Be in MODE 3.	30 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. One RTB train inoperable.	-----------------NOTE------------------ One train may be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. ---	
	Q.1 Restore train to OPERABLE status.	24 hours
	<u>OR</u>	
	Q.2 Be in MODE 3.	30 hours
R. One or more channel(s) inoperable.	R.1 Verify interlock is in required state for existing unit conditions.	1 hour
	<u>OR</u>	
	R.2 Be in MODE 3.	7 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	---NOTE--- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. ------ Perform TADOT.	62 days on a STAGGERED TEST BASIS
SR 3.3.1.5 *	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.1.6	---NOTE--- Not required to be performed until 24 hours after THERMAL POWER is $\geq$ 75% RTP. ------ Calibrate excore channels to agree with incore detector measurements.	92 EFPD
SR 3.3.1.7	---NOTE--- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. ------ Perform COT.	184 days

* [For the function •Safety Injection Input from Engineered Safety Feature Actuation System *proposed changes to the surveillance frequency will be evaluated in a future amendment. The existing Technical Specification requirement for 31 days on a staggered test basis remains in effect.*]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-----NOTE----- This Surveillance shall include verification that interlocks P-6 (for the Intermediate Range channels) and P-10 (for the Power Range channels) are in their required state for existing unit conditions. -----	
	Perform COT.	-----NOTE----- Only required when not performed within previous 184 days ----- Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate range instrumentation <u>AND</u> Four hours after reducing power below P-6 for source range instrumentation <u>AND</u> Every 184 days thereafter

(continued)

ACTIONS (continued)
SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2 *	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.2.3	-----NOTE----- Final actuation of pumps or valves not required. ----- Perform TADOT.	31 days
SR 3.3.2.4 *	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.2.5	Perform COT.+	184 days
SR 3.3.2.6	Perform SLAVE RELAY TEST.	92 days OR 18 months for only Westinghouse AR and Potter & Brumfield MDR relay types
SR 3.3.2.7	Perform COT.	31 days

(continued)

*[For the function *Turbine Trip and Feedwater Isolation - SG Water Level - High High (P-14) proposed changes to the surveillance frequency will be evaluated in a future amendment. The existing Technical Specification requirement for actuation logic test and master relay test frequency remains 31 days on a staggered test basis.*]

+{For the function o Feedwater Isolation - Tave - Low coincident with reactor trip, P-4 the proposed changes to the surveillance frequency will be evaluated in a future amendment. The existing Technical Specification requirement Channel Operational Test Frequency of 92 days remains in place.]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	C.1 -------------NOTE-------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------------	
	Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 Be in MODE 5.	60 hours
D. * One channel inoperable.	D.1 -------------NOTE-------------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ----------------------------------	
	Place channel in trip.	72 hours
	OR	
	D.2.1 Be in MODE 3.	78 hours
	AND	
	D.2.2 Be in MODE 4.	84 hours

*[For the function Auxiliary Feedwater Loss of Offsite Power proposed changes to this Condition will be evaluated in a future amendment. *The existing Technical Specification requirements for Bypass test time of 4 hours and Required Action D:1 Place channel in trip time of 6 hours and Required Action D.2.1 Be in MODE 3 in 12 hours and Action D.2.2 Be in MODE 4 in 18 hours in remains in effect*]

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One Containment Pressure channel inoperable.	E.1 -------------NOTE-------------- One additional channel may be bypassed for up to 12 hours for surveillance testing. ----------------------------------	
	Place channel in bypass.	72 hours
	OR	
	E.2.1 Be in MODE 3.	78 hours
	AND	
	E.2.2 Be in MODE 4.	84 hours
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	F.2.1 Be in MODE 3.	54 hours
	AND	
	F.2.2 Be in MODE 4.	60 hours
G. One Steam Line Isolation Manual Initiation - individual channel inoperable.	G.1 Restore channel to OPERABLE status.	48 hours
	OR	
	G.2 Declare associated steam line isolation valve inoperable.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. One train inoperable.	H.1 -------------NOTE-------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------------	
	Restore train to OPERABLE status.	24 hours
	OR	
	H.2.1 Be in MODE 3.	30 hours
	AND	
	H.2.2 Be in MODE 4.	36 hours
I. One train inoperable.	I.1 -------------NOTE-------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------------	
	Restore train to OPERABLE status.	24 hours
	OR	
	I.2 Be in MODE 3.	30 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. * One channel inoperable.	J.1 -------------NOTE-------------- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ----------------------------------	
	Place channel in trip.	72 hours
	<u>OR</u>	
	J.2 Be in MODE 3.	78 hours
K. One Main Feedwater Pumps trip channel inoperable.	K.1 Place channel in trip.	1 hour
	<u>OR</u>	
	K.2 Be in MODE 3.	7 hours
L. One channel inoperable.	L.1 -------------NOTE-------------- One channel may be bypassed for up to 2 hours for surveillance testing provided the other channel is OPERABLE. ----------------------------------	
	Be in MODE 3.	6 hours

*[For the function *Feedwater IsolationTavg - Low coincident with Reactor Trip, P-4 proposed changes to this Condition will be evaluated in a future amendment. The existing Technical Specification requirements for Bypass test time of 4 hours and Required Action J:1 Place channel in trip time of 6 hours and Required Action J.2: Be in MODE 3 in 12 hours remains in effect*]

(continued)

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Air Release and Addition Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform ACTUATION LOGIC TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.2	Perform MASTER RELAY TEST.	92 days on a STAGGERED TEST BASIS
SR 3.3.6.3	Perform SLAVE RELAY TEST.	92 days <u>OR</u> 18 months for only Westinghouse AR and Potter & Brumfield MDR relay types
SR 3.3.6.4	--------------------------------NOTE-------------------------------- Verification of setpoint is not required. --- Perform TADOT.	 18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	Perform COT.	31 days
SR 3.3.9.3	Verify each automatic valve moves to the correct position and Reactor Makeup Water pumps stop upon receipt of an actual or simulated actuation signal.	18 months
SR 3.3.9.4	------NOTE------ Only required to be performed when used to satisfy Required Action A.3 or B.3. ------------ Perform CHANNEL CHECK on the Source Range Neutron Flux Monitors.	12 hours
SR 3.3.9.5	------NOTE------ Only required to be performed when used to satisfy Required Action A.3 or B.3. ------------ Verify combined flowrates from both Reactor Makeup Water Pumps are $\leq$ the value in the COLR.	31 days
SR 3.3.9.6	------NOTE------ Only required to be performed when used to satisfy Required Action A.3 or B.3. ------------ Perform COT on the Source Range Neutron Flux Monitors.	184 days

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 247 TO RENEWED FACILITY OPERATING LICENSE NPF-35

AND

AMENDMENT NO. 240 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By application dated December 11, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073480445), as supplemented by letter dated December 18, 2008 (ADAMS Accession No. ML083570206), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2). The December 18, 2008, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 25, 2008 (73 FR 15783).

The proposed changes would revise various TS sections to allow an increase of the reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) channel logic completion times, bypass test times, allowable outage times, and surveillance testing intervals. The licensee proposed to adopt changes previously approved by the NRC staff in Westinghouse Topical Report WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS [reactor protection system] and ESFAS Test Times and Completion Times," issued October 1998, as approved by the Nuclear Regulatory Commission (NRC) in a letter dated July 15, 1998 (ADAMS No. 9808030174). Implementation of the proposed changes is in accordance with TS Task Force (TSTF) Change Traveler TSTF-418, Revision 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)." The NRC-approved TSTF-418, Revision 2, by letter dated April 2, 2003 (ADAMS No. ML030920633).

In addition, the licensee proposed to adopt changes approved by the NRC staff in WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," dated March 2003, as approved by the NRC in a letter dated December 20, 2002. Implementation of the proposed changes is in accordance with TSTF-411, Revision 1, "Surveillance Test Interval Extension for

Components of the Reactor Protection System (WCAP-15376)." The NRC approved TSTF-411, Revision 1, by letter dated August 30, 2002 (ADAMS No. ML022460347). The licensee also requested additional changes not specifically included in the above topical reports. These changes will be evaluated in a future amendment.

2.0 BACKGROUND

The Pressurized-Water Reactor Owners Group (PWROG), formerly the Westinghouse Owners Group, Technical Specifications Optimization Program (TOP) evaluated changes to surveillance test intervals (STIs) and completion times (CTs, also called allowed outage times) for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers (RTBs). The methodology evaluated increases in surveillance intervals, test and maintenance out-of-service times, and the bypassing of portions of the reactor protection system (RPS) during test and maintenance. In 1983, the PWROG submitted Westinghouse Topical Report WCAP-10271-P, "Evaluation of Surveillance Frequencies and Out-of-Service Times for the Reactor Protection Instrumentation System," which provided a methodology for justifying revisions to a plant's TSs for the RPS. The PWROG stated in WCAP-10271 that plant staff devoted significant time and effort to perform, review, document, and track surveillance activities that, in many instances, may not be necessary because of the high reliability of the equipment. Part of the justification for the changes was their anticipated small impact on plant risk.

By letter dated February 21, 1985, the NRC staff accepted WCAP-10271, including its Supplement 1, with certain conditions. In 1989, the NRC staff issued a safety evaluation report (SER) for WCAP-10271, Supplement 2, which approved similar relaxations for the ESFAS. An additional supplemental SER issued in 1990 provided consistency between RTS and ESFAS STIs and CTs. The NRC subsequently adopted the TS changes proposed in WCAP-10271 into NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 0, issued September 1992. In this regard, the licensee implemented WCAP-10271 and its supplements in license Amendments 122 and 116 dated July 19, 1994 (ADAMS Accession No. 9407270218)

After the approval of WCAP-10271 and its supplements, the PWROG submitted Westinghouse Topical Report WCAP-14333-P, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," in May 1995. WCAP-14333-P provided justification for the following TS relaxations beyond those approved in WCAP-10271:

- Increase the bypass test times and CTs for both the reactor trip system (RTS) and ESFAS solid-state and relay protection system designs for the analog channels, increase the CT from 6 hours to 72 hours and the bypass test time from 4 hours to 12 hours for the logic cabinets, master relays, and slave relays, increase the CT from 6 hours to 24 hours.

- When the logic cabinet and RTB both cause their train to be inoperable when in test or maintenance, allow bypassing of the RTB for the period of time equivalent to the bypass test time for the logic cabinets, provided that both are tested at the same time and the plant design is such that both the RTB and the logic cabinet cause their associated electrical trains to be inoperable during test or maintenance.

The NRC staff accepted WCAP-14333 by letter dated July 15, 1998. Following the approval of WCAP-14333, the PWROG submitted WCAP-15376, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," to the NRC staff on November 8, 2000. The NRC staff subsequently approved this topical report by letter dated December 20, 2002.

WCAP-15376 specifically evaluated the analog channels, logic cabinets, master relays, and RTBs. WCAP-15376 evaluated both the solid-state protection system (SSPS) and the relay protection system. WCAP-15376 provided justification for the following TS relaxations:

- Additional extension of the STIs for components of the RPS and ESFAS beyond those previously approved in WCAP-10271.
- Extension of the STI, CT, and bypass test times for the RTBs.

3.0 REGULATORY EVALUATION

3.1 Description of System

The proposed TS modifications affect the RPS (i.e., RTS and ESFAS). The RTS is designed to initiate a reactor trip when the system exceeds limits to permissible operation. The ESFAS is designed to actuate emergency systems for accidents that challenge the normal control and heat removal systems.

The RPS comprises several major functions, including nuclear and process instrumentation, logic, reactor trip, and ESFAS actuation. Instrumentation includes sensors, power supplies, signal processing, and bistable outputs and typically consists of three or four channels. Instrumentation signals (i.e., bistable outputs) feed relays that input into the logic portion of the RPS. The logic (i.e., logic cabinets) includes two redundant and independent logic blocks consisting of two trains (A and B) of RPS logic where the input coincidence for various trip functions is determined. Either logic train initiates the ESFAS function through master and slave relays.

In addition, the RPS includes actuation paths from the Train A and Train B RPS logic to the RTBs. Normally, an RTB receives its signal from its associated RPS logic train. The system has bypass breakers for when a breaker is out of service. In this configuration, the bypass breaker is associated with the logic train of the operable RTB. The RPS utilizes two normally closed RTBs and two normally open bypass breakers. Train A RPS logic actuates RTB A, and Train B logic actuates RTB B. Opening of either RTB will disconnect power from the control rods, causing a reactor trip.

Catawba 1 and 2 utilize an SSPS for the logic portion of the RPS.

3.2 Proposed TSs Changes

The licensee proposed the following revisions to the TSs as listed in Section 2 of Enclosure 1 to the December 11, 2007, application.

LCO 3.3.1, RTS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition D	• Power Range Neutron Flux-High • Power Range Neutron Flux - High Positive Rate	Existing NOTE - Bypass Test Time from 4 hours to 12 hours
		Required Action D revised to extend time before placing in tripped condition from 6 hours to 72 hours (and to extend time to be in Mode 3 from 12 hours to 78 hours) and is revised and restructured to reduce potential for confusion.
Condition E	• Power Range Neutron Flux - Low • Overtemperature Delta-T • Overpower Delta-T • Pressurizer Pressure - High • Steam Generator (SG) Water Level - Low Low	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.
		Required Action E.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action E.2: Be in Mode 3 changed from 12 hours to 78 hours
Condition F	• Intermediate Range Neutron Flux	Required Action F.1: Reduce thermal power to < P-6 changed from 2 hours to 24 hours
		Required Action F.2: Increase thermal power to > P-10 changed from 2 hours to 24 hours
Condition L	• Pressurizer Pressure - Low • Pressurizer Water Level - High • Reactor Coolant Flow - Low: Two Loops • Undervoltage RCPs • Underfrequency RCPs	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.
		Required Action L.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action L.2: Reduce thermal power to < P-7 changed from 12 hours to 78 hours
Condition M	• *Reactor Coolant Flow - Low: Single Loop*	*Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.*
		Required Action M.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action M.2: Reduce thermal power to < P-8 changed from 10 hours to 76 hours
Condition N	• Turbine Trip - Stop Valve EH Pressure Low	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.
		Required Action N.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action N.2: Reduce thermal power to < P-9 changed from 10 hours to 76 hours

LCO 3.3.1, RTS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition O	• Turbine Trip - Turbine Stop Valve Closure	Required Action O.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action O.2: Reduce thermal power to < P-9 changed from 10 hours to 76 hours
Condition P	• Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) • Automatic Trip Logic (Modes 1 and 2)	Required Action P.1: Restore train to OPERABLE status changed from 6 hours to 24 hours
		Required Action P.2: Be in Mode 3 changed from 12 hours to 30 hours
Condition Q	• Reactor Trip Breakers	Note 1: One train Bypass time changed from 2 to 4 hours
		Note 2: Deleted one RTB bypass time
		Required Action Q.1: Restore train to OPERABLE status changed from 1 hr to 24 hours
		Required Action Q.2: Be in Mode 3 changed from 7 hrs to 30 hrs
SR 3.3.1.4	• Reactor Trip Breakers • Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	TADOT frequency changed from 31 days to 62 days on a staggered test basis.
SR 3.3.1.5	• *Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)* • Low Power Reactor Trips Block, P-7 • Automatic Trip Logic	Actuation Logic Test frequency changed from 31 days to 92 days on a staggered test basis.

SR 3.3.1.7	• Power Range Neutron Flux-High
	• Power Range Neutron Flux-High Positive Rate
	• Source Range Neutron Flux (Modes 3,4, & 5)
	• Overtemperature Delta-T
	• Overpower Delta-T
	• Pressurizer Pressure
	    o Low
	    o High
	• Pressurizer Water Level - High
	• Reactor Coolant Flow - Low
	    o Single Loop
		Channel Operational Test frequency changed from 92 days to 184 days.

<table>
<tr><th colspan="3">LCO 3.3.1, RTS Instrumentation</th></tr>
<tr><th>Affected Condition</th><th>Affected Instrumentation</th><th>Proposed Change</th></tr>
<tr><td></td><td>o Two Loops
• Steam Generator (SG) Water Level - Low-Low</td><td></td></tr>
<tr><td rowspan="2">SR 3.3.1.8</td><td rowspan="2">• Power Range Neutron Flux – Low
• Intermediate Range Neutron Flux
• Source Range Neutron Flux</td><td>Note that states “Only required when not performed within previous 92 days”, frequency changed from 92 days to 184 days.</td></tr>
<tr><td>Channel Operational Test frequency changed from 92 days to 184 days.</td></tr>
</table>

<table>
<tr><th colspan="3">LCO 3.3.2, ESFAS Instrumentation</th></tr>
<tr><th>Affected Condition</th><th>Affected Instrumentation</th><th>Proposed Change</th></tr>
<tr><td rowspan="3">Condition C</td><td rowspan="3">• Automatic Actuation Logic and Actuation Relays
o Safety Injection
o Containment Spray
o Containment Isolation - Phase A
o Containment Isolation - Phase B
o Automatic Switchover to Containment Sump</td><td>Required Action C.1: Restore train to OPERABLE status changed from 6 hours to 24 hours</td></tr>
<tr><td>Required Action C.2.1: Be in Mode 3 changed from 12 hours to 30 hours</td></tr>
<tr><td>Required Action C.2.2: Be in Mode 5 changed from 42 hours to 60 hours</td></tr>
<tr><td rowspan="4">Condition D</td><td rowspan="4">• Safety Injection
o Containment Pressure - High
o Pressurizer Pressure - Low
• Steam Line Isolation
o Steam Line Pressure - Low
o Steam Line Pressure - Negative Rate High
• Feedwater Isolation
o SG Water Level – High High (P-14)
• Auxiliary Feedwater
o SG Water Level - Low Low
o Loss of Offsite Power</td><td>Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.</td></tr>
<tr><td>Required Action D.1: Place channel in trip changed from 6 hours to 72 hours</td></tr>
<tr><td>Required Action D.2.1: Be in Mode 3 changed from 12 hours to 78 hours</td></tr>
<tr><td>Required Action D.2.2: Be in Mode 4 changed from 18 hours to 84 hours</td></tr>
</table>

LCO 3.3.2, ESFAS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition E	• Containment Spray o Containment Pressure - High High • Containment Isolation - Phase B o Containment Pressure - High High • Steam Line Isolation o Containment Pressure - High High	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours. Required Action E.1: Place channel in bypass changed from 6 hours to 72 hours Required Action E.2.1: Be in Mode 3 changed from 12 hours to 78 hours Required Action E.2.2: Be in Mode 4 changed from 18 hours to 84 hours
Condition H	• Automatic Actuation Logic and Actuation Relays o Steam Line Isolation o Feedwater Isolation o Auxiliary Feedwater	Required Action H.1: Restore train to OPERABLE status changed from 6 hours to 24 hours Required Action H.2.1: Be in Mode 3 changed from 12 hours to 30 hours Required Action H.2.2: Be in Mode 4 changed from 18 hours to 36 hours
Condition I	• Turbine Trip o Automatic Actuation Logic and Actuation Relays	Required Action I.1: Restore train to OPERABLE status changed from 6 hours to 24 hours Required Action I.2: Be in Mode 3 changed from 12 hours to 30 hours
Condition J	• Turbine Trip o SG Water Level - High High (P-14) • *Feedwater Isolation* o *Tavg - Low coincident with Reactor Trip, P-4*	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours. Required Action J.1: Place channel in trip changed from 6 hours to 72 hours Required Action J.2: Be in Mode 3 changed from 12 hours to 78 hours
SR 3.3.2.2	• Automatic Actuation Logic and Actuation Relays o Safety Injection o Containment Spray o Containment Isolation - Phase A o Containment Isolation - Phase B o Steam Line Isolation o Turbine Trip and Feedwater Isolation o Auxiliary Feedwater o Automatic Switchover to Containment Sump	Actuation Logic Test frequency Changed from 31 days to 92 days on a staggered test basis.

LCO 3.3.2, ESFAS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
	• *Turbine Trip and Feedwater Isolation - SG Water Level - High High (P-14)*	
SR 3.3.2.4	• Automatic Actuation Logic and Actuation Relays o Safety Injection o Containment Spray o Containment Isolation - Phase A o Containment Isolation - Phase B o Steam Line Isolation o Turbine Trip and Feedwater Isolation o Auxiliary Feedwater o Automatic Switchover to Containment Sump • *Turbine Trip and Feedwater Isolation - SG Water Level - High High (P-14)*	Master Relay Test frequency Changed from 31 days to 92 days on a staggered test basis.
SR 3.3.2.5	• Safety Injection o Containment Pressure - High o Pressurizer Pressure - Low • Containment Spray o Containment Pressure - High High • Containment Isolation - Phase B o Containment Pressure - High High • Steam Line Isolation o Containment Pressure - High High o Steam Line Pressure - Low o Steam Line Pressure Negative Rate – High	Channel Operational Test Frequency Changed from 92 days to 184 days.

LCO 3.3.2, ESFAS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
	• Turbine Trip and Feedwater Isolation o SG Water Level - High High (P-14) o *Feedwater Isolation - Tave - Low coincident with reactor trip, P-4* • Auxiliary Feedwater o SG Water Level - Low Low • ESFAS Interlocks o Pressurizer Pressure, P-11 o Tave – Low Low, P-12	

All affected instrumentation in italics listed above will be evaluated in a future amendment.

3.3 Regulatory Requirements and Guidance

Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities.

Section 50.36(c)(3), "Technical specifications," of 10 CFR requires a licensee's TSs to have SRs for testing, calibration, and inspection to assure that the necessary quality of systems and components is maintained, that facility operations remain within safety limits, and that the Limiting Conditions of Operation will be met. Although 10 CFR 50.36 does not specify specific TS requirements, the rule implies that required actions for failure to meet the TS test bypass times, CTs, and STIs must be based on reasonable protection of the public health and safety. Therefore, the NRC staff must have reasonable assurance that the proposed TS changes will not adversely affect the performance of required safety functions in accordance with the design basis accident analysis in Chapter 15 of the licensee's updated final safety analysis report (UFSAR) with the proposed test bypass times, CTs, and STIs.

Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Maintenance Rule), of 10 CFR requires licensees to monitor the performance or condition of systems, structures, and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions. The implementation and monitoring program guidance of Section 2.3 of Regulatory Guide (RG) 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued November 2002, and Section 3 of RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued August 1998, states that monitoring performed in conformance with the Maintenance Rule can be used when it is sufficient for the

SSCs affected by the risk-informed application. In addition, 10 CFR 50.65(a)(4), as it relates to the proposed surveillance, bypass test times, and CTs, requires the assessment and management of the increase in risk that may result from the proposed maintenance activity.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for the design, fabrication, construction, testing, and performance of SSCs important to safety. In this regard, General Design Criterion (GDC) 13, "Instrumentation and Control," states that the licensee shall provide appropriate controls to maintain these variables and systems within prescribed operating ranges. Further, GDC 21, "Protection System Reliability and Testability," states that the design of the protection system shall provide for high functional reliability and inservice testability commensurate with the safety functions to be performed. The design of the protection system shall permit periodic testing of its functioning when the reactor is in operation, including the capability to test channels independently to determine failures and losses of redundancy that may have occurred.

RG 1.174 describes a risk-informed approach with associated acceptance guidelines for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights.

RG 1.177 describes an acceptable risk-informed approach and additional acceptance guidance geared toward the assessment of proposed permanent TS CT changes. RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed CT TS change, as discussed below:

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RGs 1.174 and 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out of service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Tier 1 also considers the cumulative risk of the present TS change in light of past (related) applications or additional applications under review along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed application, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that appropriate restrictions are in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and that the licensee takes appropriate compensatory measures to avoid risk-significant configurations that may not have been considered during the Tier 2 evaluation. Compared with Tier 2, Tier 3 provides additional coverage to ensure that the licensee identifies risk-significant plant equipment outage configurations in a timely manner and appropriately evaluates the risk impact of out-of-service equipment before performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (Section (a)(4)), subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The purpose of the CRMP is to ensure that the licensee will appropriately assess, from a risk perspective, equipment removed from service before or during the proposed extended CT.

RGs 1.174 and 1.177 also describe acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analyses used to support the proposed TS changes will remain valid. The monitoring program should include means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule can be used when such monitoring is sufficient for the SSCs affected by the risk-informed application.

Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (hereafter referred to as the SRP), provides general guidance for evaluating the technical basis for proposed risk-informed changes. SRP Section 16.1, "Risk-Informed Decision Making: Technical Specifications," provides more specific guidance related to risk-informed TS changes, including CT changes as part of risk-informed decision making. SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," addresses the technical adequacy of a baseline PRA used by a licensee to support license amendments for an operating reactor. SRP Section 19.2 states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following five key principles:

(1) The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

(2) The proposed change is consistent with the defense-in-depth philosophy.

(3) The proposed change maintains sufficient safety margins.

(4) When proposed changes increase CDF or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(5) The licensee should monitor the impact of the proposed change using performance measurement strategies.

4.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's analysis in support of its proposed application dated December 11, 2007, as supplemented by letter dated December 18, 2008.

4.1 Background of TS Changes as described in TSTFs

The Westinghouse Owners Group Technical Specification Optimization Program (WOG TOPS) evaluated changes to surveillance test intervals and allowed outage times (AOTs) for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers as documented in WCAP-10271-P-A series of reports. The NRC approved increasing the STIs, bypass test times, and AOTs for the analog channels, as well as the AOTs for the logic cabinets, master relays, and slave relays. A probabilistic risk assessment approach was used in these analyses which included assessing the impact of the changes on signal availability and plant safety. The justification for the acceptability of the changes was the small impact the changes had on plant safety. It was also demonstrated that increasing the surveillance test intervals for the analog channels leads to a decrease in inadvertent reactor trips since fewer test activities will be performed with a channel in trip. This provides a safety benefit.

The approach used in this program and presented in WCAP-14333-P-A Revision 1 (hereafter referred to as WCAP-14333) and WCAP-15376-P-A Revision 1 (hereafter referred to as WCAP-15376) is consistent with the approach established by WOG TOPS. This includes the fault tree models, signals, component reliability database, and most of the test and maintenance assumptions. Several changes in modeling were implemented to enhance the approach or to remove unnecessary conservatisms, such as the common cause modeling approach for analog channels and the frequency of maintenance activities. The plant-specific model used for the risk analysis was also changed. Differences in analysis methods from the WOGTOPS WCAP-10271-P-A (hereafter referred to as WCAP-10271) series of reports are discussed in Section 7.1 of WCAP-14333 and in Section 8.3.5 of WCAP-15376. In this regard, the licensee has implemented WCAP-10271 and its supplements in license Amendments Nos.122 and 116 dated July 19, 1994 (ADAMS Accession No. 9407270218)

Important to understanding the analysis and approach is a basic understanding of the RTS and ESFAS designs, and also the performance of test and maintenance activities on these systems. This information is contained in WCAP-14333.

WCAP-14333 provides the justification for increasing the bypass times for testing and the CTs in the RPS instrumentation and ESFAS instrumentation TSs. The NRC issued a safety evaluation on July 15, 1998, approving WCAP-14333.

These improvements will allow additional time to perform maintenance and test activities, enhance safety, provide additional operational flexibility, and reduce the potential for forced outages related to compliance with the RTS and ESFAS instrumentation TSs. Industry information has shown that a significant number of trips that have occurred are related to instrumentation test and maintenance activities, indicating that these activities should be completed with caution and sufficient time should be available to complete these activities in an orderly and effective manner. These changes have been incorporated in, TSTF-418, Revision 2, "RPS and ESFAS Test Times and Completion Times" (WCAP-14333).

WCAP-15376-P, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," provides the justification for the following changes to the Improved Standard Technical Specifications for the RTS Instrumentation (3.3.1) and ESFAS Instrumentation (3.3.2):

1. Increase the CT and the bypass test time for the reactor trip breakers.

2. Increase the STI for the reactor trip breakers, master relays, logic cabinets, and analog channels.

The evaluation in WCAP-15376 considers both the Solid State Protection System and the Relay Protection System.

Depending on the plant protection system design, some of the actuation logic and master relays associated with the Containment Purge and Exhaust Isolation Instrumentation (3.3.6) TS may be processed through the Relay or Solid State Protection System. Since the STIs for the actuation logic and master relays of the ESFAS Instrumentation were justified to be relaxed in this report, these STI relaxations are also applicable to the actuation logic and master relays for all signals processed through the Relay or Solid State Protection System.

The STI for the source range neutron flux CHANNEL OPERATIONAL TEST (COT) in the RTS Instrumentation (3.3.1) TS was justified to be relaxed in this report. Since this source range neutron flux channel is also used for the Boron Dilution Protection System (BDPS) in TS 3.3.9, the STI relaxation is also applicable to that STI. These changes have been incorporated in TSTF-411, Revision 1, "Surveillance Test Interval Extensions for Components of the Reactor Protection System" (WCAP-15376).

Condition F of TS 3.3.1 applies when THERMAL POWER is between the P-6 and P-10 interlock setpoints and one intermediate range channel is inoperable. The Completion Time associated with this Condition permits 2 hours to exit this power interval. NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 5, Action 3b of Table 3.3-1 allowed one Intermediate Range Neutron Flux channel to be inoperable for an indefinite period of time with power level above the P-6 (Intermediate Range Neutron Flux Interlock) but below 10% of RATED THERMAL POWER. The inoperable channel was required to be restored to OPERABLE status prior to increasing power above 10%. During the development of the NUREG, the justification for changing this action to the NUREG-1431 Condition F did not describe or provide any justification for the 2 hours.

TSTF-246, "RTS Instrumentation, 3.3.1 Condition F Completion Time," increases the Completion Times for Condition F.1 and F.2 from 2 hours to 24 hours. This TSTF was approved by the NRC on March 22, 1999.

4.2 Summary Description of the TS Changes Proposed by Licensee

The following table summarizes the proposed WCAP-14333 changes, as applicable to Catawba 1 and 2.

RPS/ESFAS Components	CT		Bypass Test Time	
	Current (Hour)	Proposed (Hour)	Current (Hour)	Proposed (Hour)
Analog Channels	6+6[1]	72+6	4	12
Logic Cabinets	6+6	24+6	4	No Change
Master Relays	6+6	24+6	4	No Change
Slave Relays	6+6	24+6	4	No Change
RTBs	1	24+6[2]	2	4[2]

1. The +6 hours is the time allowed for the specified mode change.

2. WCAP-14333 does not directly revise the RTB CT and bypass test times, and it is assumed that the bypass test times for the RTBs and the logic cabinets are separate and independent. However, WCAP-14333 assumes that with either a logic cabinet or RTB in test or maintenance their associated train is also unavailable. Based on this, the analysis presented in WCAP-14333 includes a provision to accept a bypass test time of the RTBs equivalent to the bypass test time for the logic cabinets provided that: (1) both are tested concurrently, and (2) the plant design is such that both the RTB and the logic cabinet cause their associated electrical trains to be inoperable during test or maintenance. Therefore, the RTB bypass test time is extended to 4 hours for this maintenance configuration. With the implementation of WCAP-15376, the RTB bypass test time is increased to 4 hours, consistent with the logic cabinet bypass test time.

The following table summarizes the proposed WCAP-15376 changes, as applicable to Catawba 1 and 2

RPS Component	STI		CT		Bypass Test Time	
	Current (Month)	Proposed (Month)	Current (Hour)	Proposed (Hour)	Current (Hour)	Proposed (Hour)
Logic Cabinets	2	6	No Change Requested		No Change Requested	
Master Relays	2	6				
Analog Channels	3	6				
RTBs	2	4	1	24	2	4

4.3 Review of Methodology

In accordance with SRP Sections 19.1, 19.2, and 16.1, the NRC staff reviewed the licensee's incorporation of WCAP-14333 and WCAP-15376 using the three-tiered approach and the five key principles of risk-informed decision making presented in RGs 1.174 and 1.177 and the SERs dated July 15, 1998 and December 20, 2002 respectively; conditions and limitations for WCAP-14333 and WCAP-15376.

4.4 Key Information Used in the Review

The key information used in the NRC staff's review comes from Enclosure 1 and Attachments 5 and 6 to Enclosure 1 of the application dated December 11, 2007, as supplemented December 18, 2008; TSTF-411, Revision 1, and TSTF-418, Revision 2; as approved by SERs dated August 30, 2002, and April 2, 2003, respectively; and the NRC staff's SERs on WCAP-14333 and WCAP-15376. The NRC staff also referred to previous SERs related to WCAP-10271 and the licensee's individual plant examination (IPE) and individual plant examination of external events (IPEEE) assessments.

4.5 Traditional Engineering Evaluation

The proposed changes do not involve changes to actuation setpoints, setpoint tolerance, testing acceptance criteria, or channel response times. No hardware changes are proposed or required to implement these changes at the plant. The licensee has stated that this amendment request will allow more time for maintenance and testing activities, provide additional operational flexibility, and reduce the potential for forced outages to comply with the current RTS/ESFAS instrumentation TSs. The licensee explained that industry information has shown that a significant number of reactor trips are related to instrumentation test and maintenance activities, indicating that the TSs should provide sufficient time to complete these activities in an orderly and efficient manner.

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 of the NRC staff's philosophy of risk-informed decision making, which concern compliance with current regulations, evaluation of defense in depth, evaluation of safety margins, and performance measurement strategies. Key principle 4 is evaluated in Section 4.6.1 of this SE.

With respect to key principles 1, 2, and 3, the NRC staff previously performed a generic evaluation of WCAP-14333 and WCAP-15376. The NRC staff's review of the changes found that WCAP-14333 and WCAP-15376 were consistent with the accepted guidelines of RG 1.174 and RG 1.177, and NRC staff guidance as outlined in NUREG-0800, "Standard Review Plan." From traditional engineering insights, the NRC staff found that the proposed changes continue to meet the regulations, have no impact on the defense-in-depth philosophy, and would not involve a significant reduction in the margin of safety.

With respect to key principle 5, RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS CTs, bypass test times, and surveillance intervals do not degrade operational safety over time and that no adverse degradation results from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule can be used when such monitoring is sufficient for the SSCs affected by the risk-informed application.

4.6 NRC Staff's Technical Evaluation (Probabilistic Risk Assessment)

4.6.1 Key Principle 4: Risk Evaluation

The changes proposed by the licensee employ a risk-informed approach to justify changes to CTs, bypass test times, and STIs. The risk metrics, ΔCDF, ΔLERF, ICCDP, and ICLERP, developed in the topical report and that the licensee used to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177.

To determine that WCAP-14333 and WCAP-15736 are applicable to Catawba 1 and 2, the licensee addressed the conditions and limitations of the NRC staff's SERs and the implementation guidance developed by the PWROG that compares plant-specific data to the generic analysis assumptions. The evaluation compared the general baseline assumptions, including surveillance, maintenance, calibration, actuation signals, procedures, and operator actions, to confirm that the generic evaluation assumptions used in the topical reports are also applicable to Catawba 1 and 2.

The following paragraphs discuss the licensee's evaluation of the SER conditions and limitations of WCAP-14333 and WCAP-15376.

(1) A licensee should confirm the applicability of the WCAP-14333 and WCAP-15376 analyses for its plant.

The WCAP-15376 estimates for LERF were based on the reference plant having a large dry containment and the assumption that the only contributions to LERF would be from containment bypass or core damage events when the containment is not isolated. The NRC staff SER stated there may be exceptions to this assumption, including plants with ice condenser containments. Therefore, a plant-specific assessment of containment failures should be performed to determine if there are any impacts on the proposed TS change.

Catawba 1 and 2 utilizes an ice condenser containment. The licensee stated that LERF for Catawba 1 and 2 is dominated by intersystem LOCAs (ISLOCA) and station black out (SBO) sequences. The ice condenser containment depends on a hydrogen igniter system to control hydrogen during severe accidents, which would be unavailable during SBO events due to unavailability of electric power. The proposed RTS and ESFAS instrumentation CT, bypass test times, and STIs would therefore not impact SBO sequences which results in LERF.

To supplement the qualitative LERF evaluation, the licensee estimated the ΔLERF by comparing the internal events CDF and LERF to the estimated CDF and LERF results with both WCAP-14333 and WCAP-15376 implemented for the generic plant. The licensee estimated the ΔLERF using WCAP-15376 cumulative CDF estimates (pre-TOP to WCAP-15376) and a bounding plant-specific LERF/CDF ration. Both estimates are below RG. 1.174 acceptance of 1.0E-7 for a very small change. Based on qualitative and quantitative LERF evaluations, the licensee found the plant specific LERF contribution is consistent with WCAP-15376 assumptions.

Therefore, based on the evaluation presented in Section 4.6.2, Tier 1, of this SE, the NRC staff considers the condition satisfied for Catawba 1 and 2.

(2) Under WCAP-14333 and WCAP-15376, the licensee should address the Tier 2 and Tier 3 analyses, including risk significant configuration insights, by confirming that these insights are incorporated into its CRMP decision-making process before taking equipment out of service.

Based on the evaluation presented in Section 4.6.3 (Tier 2) and Section 4.6.4 (Tier 3) of this SE, the licensee addressed both Tier 2 and Tier 3 risk significant configurations and confirmed these insights are incorporated into the Catawba 1 and 2 CRMP. Therefore, the NRC staff considers this condition satisfied for Catawba 1 and 2.

(3) The licensee should evaluate the risk impact of concurrent testing of one logic cabinet and associated RTB on a plant-specific basis to ensure conformance with the WCAP-15376 evaluation, including the guidance of RGs 1.174 and 1.177.

The licensee showed that the generic analysis presented in WCAP-15376 is applicable to Catawba 1 and 2. WCAP-15376 did not specifically evaluate or preclude concurrent testing of one logic cabinet and associated RTB. Based on this, the NRC staff questioned the applicability of the topical report to this particular maintenance configuration. In response to an NRC staff RAI on WCAP-15376, the PWROG provided generic risk estimates that assumed concurrent testing. The resulting ICCDP estimate was higher than the WCAP-15376 results but within the acceptance guidelines of RG 1.177. Based on the applicability of WCAP-15376 to Catawba 1 and 2, and an ICCDP estimate within the acceptance guidelines of RG 1.177, the NRC staff considers Condition 3 to be satisfied.

(4) To ensure consistency with the reference plant, the licensee should confirm that the model assumptions for human reliability in WCAP-15376 are applicable to the plant-specific configuration.

Enclosure 1, Attachment 6A Table 5 of the licensee's submittal confirmed that the assumptions regarding human reliability used in WCAP-15376 are applicable to Catawba 1 and 2. The licensee's review concluded that for the operator actions identified in WCAP-15376 and credited in the Catawba PRA, plant procedures, training and sufficient time are available consistent with the assumptions in WCAP-15376. Based on the above, the NRC staff considers Condition 4 to be satisfied.

(5) For future digital upgrades with increased scope, integration, and architectural differences, the NRC staff finds that the generic applicability of WCAP-15376 to a future digital system is not clear and should be considered on a plant-specific basis. Catawba 1 and 2 design is based on the SSPS, therefore this condition is not applicable to the implementation of WCAP-15376 at Catawba 1 and 2.

(6) WCAP-15376 included an additional condition based on the PWROG commitment that each plant review its plant-specific setpoint calculation methodology to ensure that the extended STIs do not adversely impact the plant-specific setpoint calculations and assumptions for instrumentation associated with the extended STIs.

The additional condition requires that the licensees are to perform plant-specific reviews of RPS and ESFAS setpoint uncertainty calculations and assumptions, including instrument drift, to determine the impact of extending the surveillance frequency of the COT from 92 days to 184 days. Catawba 1 and 2 have performed this plant specific evaluation and the results are summarized below.

The licensee reviewed the plant specific RTS and ESFAS setpoint uncertainty calculations and assumptions, including instrument drift, to determine the impact of extending the COT surveillance from 92 days to 184 days and determined that the values used in the Catawba 1 and 2 setpoint studies properly accounted for drift due to the extended STIs. Based on their review of the setpoint uncertainty calculations, the licensee stated that they do not anticipate any impact on setpoint uncertainty due to extending the STIs from 92 days to 184 days. However, the licensee committed to trend and evaluate as-found and as-left data for the three representative trip functions analyzed in WCAP-15376 (i.e., over-temperature ΔT (OTDT), steam generator (SG) level, and pressurizer pressure) for 2 years (4 data points) following implementation of the proposed changes.

The NRC staff review found the proposed change to extend COT surveillance from 92 days to 184 days in Catawba 1 and 2 TSs sections 3.3.1, 3.3.2 within the scope of the NRC staff's SE on WCAP-15376 and, therefore, acceptable. Further to this, the NRC staff finds that the licensee's regulatory commitments within 90 days of NRC approval of the LAR, will ensure SSPS availability.

Based on the above, the NRC staff concludes that the proposed changes are acceptable and the licensee continues to meet the requirements of 10 CFR 50.36 for setpoints.

4.6.2 Tier 1: Probabilistic Risk Assessment Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk based on the implementation of WCAP-14333 and WCAP-15376 at Catawba 1 and 2. The Tier 1 NRC staff's review involves: (1) evaluation of the technical adequacy of the PRA and its application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Technical Adequacy

WCAP-14333 and WCAP-15376 provided a generic PRA model for the evaluation of the CT, test bypass time and STI extensions. Although the WCAP-14333 and WCAP-15376 SERs accepted the use of a representative model as generally reasonable, the application of the representative model and the associated results to a specific plant introduces a degree of uncertainty because of modeling, design, and operational differences. Therefore, each licensee adopting WCAP-14333 and WCAP-15376 will need to confirm that the topical report analyses and results are applicable to its plant.

The NRC staff reviewed the information provided in the proposed application and the findings and conditions of the NRC staff's WCAP-14333 and WCAP-15376 SERs. WCAP-14333 and WCAP-15376 do not require the use of the Catawba 1 and 2 PRA or plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP in the implementation of either topical report. However, in its SER for WCAP-14333 and WCAP-15376, the NRC staff found that the applicability of the generic PRA analysis for the proposed CT, bypass test time, and STI changes to other Westinghouse plants may not be representative based on design variations in actuated systems and the contribution to plant risk from accident classes impacted by the proposed change. The licensee reviewed the scope and detail of the Catawba 1 and 2 PRA using the representative topical report PRA parameters to demonstrate the plant-specific applicability of the proposed CT, bypass test times, and STI changes. The licensee compared actuation logic; component test, maintenance, and calibration times/intervals; at-power maintenance; ATWS; total internal events CDF; transient events; operator actions; RTS trip actuation signals; and ESFAS actuation signals to plant-specific values. Based on the comparison to the implementation guidelines and NRC staff's SER conditions and limitations for WCAP-14333 and WCAP-15376, the NRC staff concluded that WCAP-14333 and WCAP-15376 are applicable to Catawba 1 and 2.

Peer Review

The original IPE for Catawba 1 and 2 was submitted in 1992, Revision 3 was issued in December 2004, and Revision 3a was completed December 2005.

As stated by the licensee, the following is a list of the reviews conducted on the PRA modeling which assures the technical adequacy of the existing PRA:

- A peer review sponsored by the Electric Power Research Institute (EPRI) was conducted on the original Catawba PRA dated August 18, 1987.
- An NRC SER was issued on the IPE and IPEEE for Catawba 1 and 2 June 7, 1994 and April 12, 1999 respectively.
- In March 2002, a peer review of the Catawba 1 and 2 PRA was conducted as part of the Westinghouse Owners Group PRA Certification Program.
- In August 2008, a PRA Technical Adequacy Self-Assessment was conducted against the Supporting Requirements (SRs) in the ASME standard (American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power

Plant Applications," ASME-RA-Sc-2007) and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," for Catawba 1 and 2.

The licensee did not identify any plant-specific design or operability issue that would invalidate the generic results. Based on its review, the NRC staff concludes that the generic results are applicable to Catawba 1 and 2.

PRA Results and Insights

Cumulative Risk

WCAP-15376 evaluated the cumulative CDF risk from pre-TOP (WCAP-10271 not incorporated) to WCAP-15376 implementation. For this case, the cumulative impact on the CDF for 2-out-of-4 logic was within the RG 1.174 acceptance guidelines of less than 1E-6/year, representing a very small change. The cumulative impact on CDF for 2-out-of-3 logic was slightly above the RG 1.174 acceptance guideline for a very small change, but within the acceptance guidelines for a small change. For Catawba 1 and 2, the cumulative risk is limited from the TOP condition (WCAP-10271 incorporated in plant licensing) to WCAP-15376 implementation. The NRC staff finds the proposed change for Catawba 1 and 2 is from TOP to WCAP-15376, the change in cumulative risk is expected to be less than the WCAP-15376 estimates.

The licensee evaluated plant-specific design or operational modifications that are not reflected in the Catawba 1 and 2 PRA. In the December 11, 2007, application the licensee confirmed that there have not been any modifications to the RTB or ESFAS that impact the proposed implementation of WCAP-14333 and WCAP-15376.

External Events

In the SER for WCAP-14333, the risk impact from external events was qualitatively considered for fires and seismic events. The NRC staff concluded that the proposed changes will have only a very small impact on the risk from external events. The licensee also evaluated the proposed WCAP-14333 and WCAP-15376 RPS and ESFAS CTs, test bypass times, and STIs for their potential impact on external events including fire, seismic, and high winds, floods, and other (HFO) events for Catawba 1 and 2. The proposed changes will increase the unavailability of the affected SSC by increasing the CT for the analog cabinets, logic cabinets, master relays, slave relays, and RTBs. To be important for an external event, the external event must occur while the SSC is in the extended CT. Based on the initial low risk from these external events and the small increase in unavailability, the NRC staff concludes the change in risk and the ICCDP should remain very small and would not cause the RG 1.174 and RG 1.177 acceptance guidance to be exceeded. The following paragraphs discuss the contribution to total risk for these events.

Fires

As stated in the licensee's December 18, 2008, supplemental letter,

> The risk significant fire initiating events in the PRA are those fires that result in failure of the component cooling water (KC) system. This was initially noted in the IPEEE report submitted to the NRC on June 21, 1994 (Section 1.4.1 of IPEEE). The component cooling system provides cooling to the pumps for the majority of the mitigating systems. Consequently, when component cooling is failed by a fire, all of the significant mitigating systems are failed.
>
> The dominant fire sequences included a loss of Reactor Coolant Pump (RCP) seal cooling support systems leading to RCP seal LOCAs. In these sequences, core damage can be mitigated by operator actions to manually start the Standby Shutdown Facility (SSF).
>
> The fire accident sequences in the CDF quantification that included both internal and external events were reviewed to identify sequences that contain failures of ESFAS actuation signals or failures of the RTS.

The results are shown in the table provided by the licensee below.

Fire +

Initiator Name*	Description	CDF for ESFAS/RTS Actuation Signal Failures in Fire Sequences	Fire Event CDF	Percent Contribution to Fire Event CDF
%FCBLR	Cable Room Fire Causes A Loss Of Component Cooling (KC) Water	0/yr	9.4E-7/yr	0.0%
%FKC	KC Power Cable Initiating Event	0/yr	3.6E-07/yr	0.0%
%FCR	Control Room Fire Causes A Loss Of KC	0/yr	2.0E-07/yr	0.0%
	Total Fire	0/yr	1.5E-06/yr	0.0%

*Other fire initiating events are included in the PRA model, but only the ones listed in Table 2 appear in the cut set file. +Applies to both Units 1 and 2.

The ESFAS/RTS failures are not a dominant contributor for fire events. The review indicated that none of the fire sequences contain these types of failures. Therefore, the NRC staff found the small increases in signal unavailability due to the proposed TS changes will have a very small impact on the fire external event CDF and will not impact the conclusions made for the proposed RTS and ESFAS extended STIs and CTs.

Seismic Events

The NRC staff's SER for WCAP-14333 concluded that for plants adopting WCAP-14333 the proposed CT and bypass test times would have a very small impact on external event risk including seismic. The instrumentation STI and RTB CT and bypass test time extensions proposed by WCAP-15376 are also expected to have a very small impact on seismic event risk.

As stated by the licensee,

> The dominant events in the current seismic PRA sequences involve either a loss of power or control with a loss of Secondary Side Heat Removal (SSHR). A majority of the sequences involve a loss of off-site power with corresponding diesel hardware or circuitry failures.
>
> ESFAS failures are explicitly modeled in the seismic PRA. The accident sequences from the Core Damage Frequency (CDF) quantification for seismic events were reviewed to identify sequences that contain failures of the Engineered Safety Features Actuation System (ESFAS).

The results are provided in the table provided by licensee below.

Seismic +

Initiator Name	Description	CDF for ESFAS Actuation Signal Failures in Seismic Sequences	Total Seismic CDF	Percent Contribution to Seismic CDF
%SEISMIC	Seismic Initiator	1.2E-07/yr	1.2E-05/yr	1.0%

+Applies to both Units 1 and 2.

The ESFAS failures are not a dominant contributor to the seismic PRA results. Seismic cut sets containing failures of ESFAS components compose approximately 1 percent of the seismic CDF. This represents a very small contribution to the CDF. The RTS signal failures are not modeled in the seismic PRA because the primary contributors of an ATWS event to the CDF are from internal transient events such as loss of main feedwater, and loss of load/turbine trip. Seismic events are relatively infrequent events that have an initiating event frequency significantly less than internal transient events. Excluding the ATWS contribution to the CDF from a seismic event is acceptable based on the very small contribution to the CDF from this event. Therefore, based on the above quantitative and qualitative evaluation of the seismic risk contribution from the RTS and ESFAS, the NRC staff finds any small increases in signal unavailability due to the proposed TS changes will have a very small impact on the seismic external event CDF and will not impact the conclusions made for the proposed RTS and ESFAS extended STIs and CTs.

High Winds, Floods, and Other External Events

As stated by the licensee,

> The effects of tornados are included in the Catawba PRA model. Dominant tornado sequences are those that induce a Loss of Offsite Power (LOOP) followed by failures of the emergency power system. Emergency power system failures are dominated by failures of the emergency diesel generators to run or common cause failures of the diesels to run.
>
> The tornado accident sequences in the CDF quantification that included both internal and external events were reviewed to identify sequences that contain failures of ESF actuation signals or failures of the RTS.

The results are shown in the table provide by the licensee below.

Tornados +

Initiator Name	Description	CDF for ESFAS/RTS Actuation Signal Failures in Tornado Sequences	Tornado Event CDF	Percent Contribution to Tornado Event CDF
%TORNSW	Tornado Causes Loss of Offsite Power (LOOP)	0/yr	3.4E-07/yr	0.0%

+Applies to both Units 1 and 2.

The ESFAS/RTS failures are not a dominant contributor for tornado events. None of the tornado sequences contain these types of failures. The evaluation conducted for the Catawba 1 and 2 IPEEE concluded that the contribution to plant risk from external flooding, transportation, and nearby facility accidents is not significant.

Therefore, the NRC staff finds based on the above quantitative and qualitative evaluation of the tornado risk and the IPEEE results for other events, any small increases in signal unavailability due to the proposed TS changes will have a very small impact on the external events CDF and will not impact the conclusions made for the proposed RTS and ESFAS extended STIs and CTs.

Total Risk Contribution

Catawba 1 and 2 has a full scope PRA where both internal and external events are modeled. The total CDF for internal and external events is 2.81E-05/year. The NRC staff finds that the base CDF of 1E-4/year will not be exceeded with the implementation of WCAP-14333 and WCAP-15376.

4.6.3 Tier 2: Avoidance of Risk-Significant Plant Configurations

A licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

Based on WCAP-14333, WCAP-15376, and the licensee's evaluations, including the functional units not evaluated generically by WCAP-14333, the licensee identified the following Tier 2 conditions as regulatory commitments:

For WCAP-14333:

- To preserve ATWS mitigation capability, activities that degrade the ability of the AFW system, reactor coolant system (RCS) pressure relief systems (pressurizer power operated relief valves (PORVS) and safety valves), ATWS mitigation system actuation circuitry (AMSAC), or turbine trip should not be scheduled when a logic train is inoperable for maintenance.

- To preserve LOCA mitigation capability, one complete emergency core cooling system train that can be actuated automatically must be maintained when a logic train is inoperable for maintenance.

- To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable for maintenance.

- Activities in electrical systems (e.g., AC and DC power) and cooling systems (e.g., essential service water and component cooling water) that support the systems or functions listed in the first three bullets should not be scheduled when a logic train is inoperable for maintenance. That is, one complete train of a function noted above must be available.

- To preserve capabilities to prevent large early releases, activities that degrade the ability of the containment spray systems, air return fans, and ice condenser should not be scheduled when a logic train is inoperable for maintenance.

For WCAP-15376:

- The probability of failing to trip the reactor on demand will increase when an RTB train is removed from service; therefore, systems designed for mitigating an ATWS event should be maintained and available. Reactor coolant system (RCS) pressure relief (pressurizer PORVS) and safety valves, AFW flow (for RCS heat removal), AMSAC, or turbine trip should not be scheduled when an RTB is inoperable for maintenance.

- Due to the increased dependence on the available reactor trip train when one logic train or RTB train is inoperable for maintenance, activities that cause master relays or slave relays in the available train to be unavailable and activities that cause analog channels to be unavailable should not be scheduled when an RTB is inoperable for maintenance.

- Activities in electrical systems (e.g., AC and DC power) that support the systems or functions listed in the first two bullets should not be scheduled when an RTB is unavailable.

The licensee evaluated concurrent component outage configurations and confirmed the applicability of the Tier 2 restrictions for Catawba 1 and 2. Based on the above, the NRC staff finds the licensee's Tier 2 analysis supports the implementation of WCAP-14333 and WCAP-15376 at Catawba 1 and 2 and satisfies the condition of the NRC staff's SERs for WCAP-14333 and WCAP-15376 regarding Tier 2.

4.6.4 Tier 3: Risk-Informed Configuration Risk Management Program

Risk assessment of online configurations for Catawba 1 and 2 is controlled under plant procedures and nuclear system directives to determine the risk significance for equipment outage configurations. The requirements contained in the procedures and directives are applicable in all plant modes of operation. The requirements control the safety impact of the combinations of equipment removed from service. The requirements also assure that the risk associated with maintenance of equipment with various plant configurations planned during at power or shutdown conditions is assessed and evaluated prior to entry into these configurations and is appropriately managed. The maintenance plan documents the allowable combinations of systems and component groups that can be worked simultaneously online or during shutdown. Work is scheduled based on established maintenance and outage periods including established maintenance frequencies and are designed to minimize on-line maintenance risk. Corrective maintenance is also evaluated with respect to surveillance and preventive maintenance activities.

A risk assessment is performed prior to work being performed and includes emergent work activities. The Catawba 1 and 2 risk assessment guidelines use the results of the Catawba 1 and 2 PRA and also consider TSs, weather and offsite power conditions. If the risk of performing a maintenance activity cannot be determined through the work control process, an additional risk assessment is performed.

The NRC staff finds that the licensee's program to control risk is capable of adequately assessing the activities being performed to ensure that high-risk plant configurations do not occur and/or compensatory actions are implemented if a high-risk plant configuration or condition should occur. As such, the licensee's program provides for the assessment and management of increased risk during maintenance activities as required by the Maintenance Rule (Section (a)(4)) and satisfies the RG 1.177 guidelines for a CRMP for the proposed change.

4.6.5 Implementation and Monitoring Program

RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS STI, CT, or bypass test times do not degrade operational safety over time and that no adverse effects occur from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. In addition, the application of the three-tiered approach in evaluating the proposed CT and bypass test times provides additional assurance that the changes will not significantly impact the key principle of defense-in-depth.

The licensee monitors the reliability and availability of the RTS and ESFAS instrumentation under 10 CFR 50.65 the Maintenance Rule. The licensee has established RTS and ESFAS performance criteria including component level criteria for the SSPS trains. The licensee component level criteria will be revised to reflect the reliability assumptions in the topical reports. The unavailability assumptions of the SSPS and RTBs were found to be within the topical report assumptions. Based on its review, the NRC staff finds Catawba 1 and 2 satisfy the RG 1.174 and RG 1.177 guidelines for an implementation and monitoring program for the proposed change.

4.7 Comparison with Regulatory Guidance

The proposed changes conform to TSTF-411, Revision 1, and the analysis performed in WCAP-15376, as approved by the NRC staff, including limitations and conditions identified in the NRC staff's SERs. Additional proposed changes conform to TSTF-418, Revision 2, and the analysis performed in WCAP-14333, as approved by the NRC staff, including limitations and conditions identified in the NRC staff's SER. As such, the NRC staff finds the implementation of WCAP-14333 and WCAP-15376 at Catawba 1 and 2 is within the RG 1.174 and RG 1.177 acceptance guidance for ΔCDF, ΔLERF, ICCDP, and ICLERP.

4.8 Deviations from Approved TSTF Changes

The licensee also requested additional changes not specifically included in the above TSTFs. These changes will be evaluated in a future amendment.

4.9 NRC Staff's Findings and Conditions

The NRC staff finds that the licensee has demonstrated the applicability of WCAP-14333 and WCAP-15376 to Catawba 1 and 2 and has met the limitations and conditions as outlined in the NRC staff's SERs. The NRC staff found the risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP as estimated by WCAP-14333 and WCAP-15376 to be applicable to Catawba 1 and 2 and within the acceptance guidelines for RG 1.174 and RG 1.177. The licensee's Tier 2 analysis evaluated concurrent outage configurations and confirmed the applicability of the risk-significant configurations identified by the NRC staff's SER limitations and conditions and topical report analysis to ensure control of these configurations. The licensee's Tier 3 CRMP is consistent with the RG 1.177 CRMP guidelines and the Maintenance Rule (Section (a)(4)) for the implementation of WCAP-14333 and WCAP-15376. The licensee monitors the reliability and availability of the RTS and ESFAS components under the Maintenance Rule (Section (a)(1)). Therefore, the NRC staff finds the TS revisions proposed by the licensee are consistent with the CTs, bypass test times, and STIs approved for WCAP-14333 and WCAP-15376 and meet the SER conditions and limitations for WCAP-14333 and WCAP-15376.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 (73 FR 15783). 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.

7.0 CONCLUSION

The Commission 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.

Principal Contributors: Clifford Doutt
John Stang
Andrew Howe
Subinoy Mazumdar

Date: December 22, 2008

Reactor Trip Breaker Test and Completion Times," issued March 2003, as approved by NRC letter datedDecember 20, 2002. Implementation of the proposed changes is consistent with TSTF Traveler # TSTF-411, Revision 1, "Surveillance Test Interval Extension för Components of the Reactor Protection System (WCAP-15376)." The NRC approved TSTF-411, Revision 1, by letter dated August 30, 2002. The licensee also requested additional changes not specifically included in the above topical reports. These changes will be evaluated in a future amendment.

Date of issuance: December 22, 2008

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance.

Amendment Nos.: 247 and 240

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the licenses and the technical specifications.

Date of initial notice in FEDERAL REGISTER: March 25, 2008 (73 FR 15783). The supplement dated December 18, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 22, 2008.

No significant hazards consideration comments received: No

DISTRIBUTION:
Nonpublic
LPL2-1 R/F
RidsNrrDorlLpl2-1 Resource
RidsNrrPMJStang Resource (hard copy)
RidsNrrLAMO'Brien Resource (hard copy)

ADAMS Accession No. ML083470177

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	OGC	NRR/LPL2-1/BC	NRR/LPL2-1/PM
NAME	JStang	MO'Brien	JBielecki	MWong	JStang
DATE	12/22/08	12/22/08	12/22/08	12/23/08	12/23/08

OFFICIAL RECORD COPY