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

December 30, 2008

Mr. Bruce H. Hamilton
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REACTOR TRIP SYSTEM AND ENGINEERGED SAFETY FEATURES ACTUATION SYSTEM COMPLETION TIMES, BYPASS TEST TIMES AND SURVEILLANCE TEST INTERVALS (TAC NOS. MD7720 AND MD7721)

Dear Mr. Hamilton:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 248 to Renewed Facility Operating License NPF-9 and Amendment No. 228 to Renewed Facility Operating License NPF-17 for the McGuire 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 TS 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, and 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

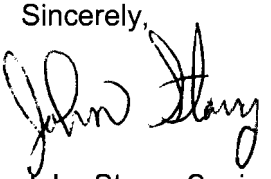
The proposed license amendment request (LAR) adopts changes as described in Westinghouse Commercial Atomic Power (WCAP) Topical Report WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the Reactor Protection System [RPS] and Engineered Safety Features Actuation System Test Times and Completion Times," issued October 1998 and approved by the NRC by 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 the NRC by 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, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 248 to NPF-9
2. Amendment No. 228 to NPF-17
3. Safety Evaluation

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 248
Renewed License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9, filed by the Duke Energy Carolinas, LLC (licensee), 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.

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-9 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248, are hereby incorporated into this license. The licensee 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

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-9
and the Technical Specifications

Date of Issuance: December 30, 2008

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 228
Renewed License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. NPF-17, filed by the Duke Energy Carolinas, LLC (the licensee), dated December 11, 2007, as supplemented December 18, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, a-s amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by 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-17 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 228, are hereby incorporated into this license. The licensee 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

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-17
and the Technical Specifications

Date of Issuance: December 30, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 248

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 228

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

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-9 page 3	NPF-9 page 3
NPF-17 page 3	NPF-17 page 3
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
3.3.2-8	3.3.2-8

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and;

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248 , are hereby incorporated into this renewed operating license. The licensee 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 June 12, 2021, 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) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operation of McGuire Nuclear Station, Units 1 and 2; and,

(6) Pursuant to the Act and 10 CFR Parts 30 and 40, to receive, possess and process for release or transfer such byproduct material as may be produced by the Duke Training and Technology Center.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at a reactor core full steady state power level of 3411 megawatts thermal (100%).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 228 , are hereby incorporated into this renewed operating license. The licensee 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 March 3, 2023, 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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 12 hours for surveillance testing and setpoint adjustment. ----------	
	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---------- One channel may be bypassed for up to 12 hours for surveillance testing. ----------	
	E.1 Place channel in trip.	72 hours
	OR	
	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
	OR	
	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.	----------NOTE---------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ----------	
	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1 Restore channel(s) to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 12 hours for surveillance testing. ----------	
	M.1 Place channel in trip.	72 hours
	OR	
	M.2 Reduce THERMAL POWER to < P-7.	78 hours
N. One Reactor Coolant Flow - Low (Single Loop) channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	N.1 Place channel in trip.	6 hours
	OR	
	N.2 Reduce THERMAL POWER to < P-8.	10 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One Turbine Trip - Low Fluid Oil Pressure channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 12 hours for surveillance testing. ----------	
	O.1 Place channel in trip.	72 hours
	OR	
	O.2 Reduce THERMAL POWER to < P-8.	76 hours
P. One or more Turbine Trip - Turbine Stop Valve Closure channels inoperable.	P.1 Place channel(s) in trip.	72 hours
	OR	
	P.2 Reduce THERMAL POWER to < P-8.	76 hours
Q. One 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
	OR	
	Q.2 Be in MODE 3.	30 hours

(continued)

ACTIONS (continued)

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	----------NOTES---------- 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	----------NOTES---------- 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	----------NOTES---------- 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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	----------NOTES---------- 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 AND Four hours after reducing power below P-10 for power and intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 184 days thereafter

(continued)

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---------- One channel may be bypassed for up to 12 hours for surveillance testing. ---------- 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

(continued)

*[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 remain in effect*]

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---------- One channel may be bypassed for up to 12 hours for surveillance testing. ---------- Place channel in trip.	72 hours
	OR	
	J.2 Be in MODE 3.	78 hours
K. One Main Feedwater Pumps trip channel inoperable.	K.1 Place channel in trip.	1 hours
	OR	
	K.2 Be in MODE 3.	7 hours
L. One required channel in one train of Doghouse Water Level-High High inoperable.	L.1 Restore the inoperable train to OPERABLE status.	72 hours
	OR	
	L.2 Perform continuous monitoring of Doghouse water level.	73 hours
M. Two trains of Doghouse Water Level-High High inoperable.	M.1 Perform continuous monitoring of Doghouse water level..	1 hour

(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	Perform COT.	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
SR 3.3.2.7	----------NOTE---------- Verification of setpoint not required for manual initiation functions. --- Perform TADOT.	18 months

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 248 TO RENEWED FACILITY OPERATING LICENSE NPF-9

AND

AMENDMENT NO. 228 TO RENEWED FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

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 McGuire Nuclear Station, Units 1 and 2 (McGuire 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 U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 25, 2008 (72 FR 15783).

The proposed changes would revise various TS sections to allow an increase in 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 described in Westinghouse Commercial Atomic Power (WCAP) 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 (ADAMS Legacy Library Accession No. 9811100114) and approved by the NRC staff in a letter dated July 15, 1998 (ADAMS Legacy Library Accession 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)" issued March 3, 2003 (ML030650848) and approved by the NRC staff in a letter dated April 2, 2003 (ADAMS Accession No. ML030920633).

In addition, the licensee proposed to adopt changes 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 19, 2003 (ADAMS Accession No. ML030870033) and approved by the NRC staff in a letter dated December 20, 2002 (ML023540534). 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)" issued August 9, 2002 (ADAMS Accession No. ML022460447) and approved by the NRC staff in a 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 REGULATORY EVALUATION

The Pressurized-Water Reactor Owners Group (PWROG), formerly the Westinghouse Owners Group (WOG), Technical Specifications Optimization Program (TOP) evaluated changes to surveillance test intervals (STIs) and completion times (CTs, also called allowed outage times (AOTs) 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 RPS during test and maintenance. In 1983, the PWROG submitted Westinghouse 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 (SE) for WCAP-10271, Supplement 2, which approved similar relaxations for the ESFAS. An additional supplemental SE 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 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 RTS and ESFAS solid-state protection system (SSPS) 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 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.

McGuire 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 (organized according to Limiting Condition of Operation (LCO)).

The TS changes for McGuire are as follows:

LCO 3.3.1, RTS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**
Condition D	• Power Range Neutron Flux-High • Power Range Neutron Flux Rate-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 M	• 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 M.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action M.2: Reduce thermal power to < P-7 changed from 12 hours to 78 hours
Condition N	• *Reactor Coolant Flow - Low: Single Loop*	*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-8 changed from 10 hours to 76 hours

Condition O	• Turbine Trip - Low Fluid Oil Pressure	Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.
		Required Action O.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action O.2: Reduce thermal power to < P-8 changed from 10 hours to 76 hours
Condition P	• Turbine Trip - Turbine Stop Valve Closure	Required Action P.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action P.2: Reduce thermal power to < P-8 changed from 10 hours to 76 hours
Condition Q	• Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) • Automatic Trip Logic (Modes 1 and 2)	Required Action Q.1: Restore train to OPERABLE status changed from 6 hours to 24 hours
		Required Action Q.2: Be in Mode 3 changed from 12 hours to 30 hours
Condition R	• Reactor Trip Breakers	Note 1: One train bypass time from 2 to 4 hours
		Note 2: One RTB bypass time deleted
		Required Action R.1: Restore train to OPERABLE status changed from 1 hr to 24 hours
		Required Action R.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 Rate - 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
		Channel Operational Test frequency changed from 92 days to 184 days.

	• Reactor Coolant Flow - Low
	  o Single Loop
	  o Two Loops
	• Steam Generator (SG) Water Level – Low Low	

SR 3.3.1.8	• Power Range Neutron Flux - Low • Intermediate Range Neutron Flux • Source Range Neutron Flux (Mode 2)	Note that states "Only required when not performed within previous 92 days", frequency changed from 92 days to 184 days. Channel Operational Test frequency changed from 92 days to 184 days.

LCO 3.3.2, ESFAS Instrumentation		
Affected Condition	**Affected Instrumentation**	**Proposed Change**

Condition C	• Automatic Actuation Logic and Actuation Relays
	  o Safety Injection
	  o Containment Spray
	  o Containment Isolation - Phase A
	  o Containment Isolation - Phase B
		Required Action C.1: Restore train to OPERABLE status changed from 6 hours to 24 hours
		Required Action C.2.1: Be in Mode 3 changed from 12 hours to 30 hours
		Required Action C.2.2: Be in Mode 5 changed from 42 hours to 60 hours

Condition D	• Safety Injection
	  o Containment Pressure - High
	  o Pressurizer Pressure – Low 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 *Station Blackout*
	  ▪ *Loss of Voltage*
	  ▪ *Degraded Voltage*
		Existing NOTE - Bypass Test Time changed from 4 hours to 12 hours.
		Required Action D.1: Place channel in trip changed from 6 hours to 72 hours
		Required Action D.2.1: Be in Mode 3 changed from 12 hours to 78 hours
		Required Action D.2.2: Be in Mode 4 changed from 18 hours to 84 hours

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 and Feedwater Isolation o Turbine Trip - SG Water Level - High High (P-14) o Feedwater Isolation - 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 • Turbine Trip and Feedwater Isolation - SG Water Level - High High (P-14)	Actuation Logic Test frequency changed from 31 days to 92 days on a staggered test basis.

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 • 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 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 • 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	Channel Operational Test frequency changed from 92 days to 184 days.

Note: All affected instrumentation listed in the tables above and marked in italics will be evaluated in a future amendment.

3.3 Regulatory Requirements and Guidance

Part 50 of 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 LCOs will be met. Although 10 CFR 50.36 does not specify detailed 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 contained in Chapter 15 of the licensee's final safety analysis report (FSAR) 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 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 "Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges." Further, GDC 21, "Protection System Reliability and Testability," states that "The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed" and "The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a 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 (i.e. Section (a)(4)), subject to the guidance provided in RG 1.177, Section 2.3.7.1, "Configuration Risk Management Program (CRMP)," 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.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (the SRP) provides guidance for evaluating risk-informed changes. Section 16.1 of the SRP, "Risk-Informed Decision Making: Technical Specifications," provides specific guidance related to risk-informed TS changes, including CT changes as part of risk-informed decision making. Section 19.1 of the SRP, "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. Section 19.2 of the SRP, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," provides guidance to reviewers 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, i.e., a specific exemption under 10 CFR 50.12.

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

(3) The proposed change maintains sufficient safety margins.

(4) When proposed changes result in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (60 FR 42622).

(5) The impact of the proposed change should be monitored 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 WOG TOP evaluated changes to surveillance test intervals and AOTs for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers as documented in the WCAP-10271-P-A series of reports. The NRC approved increasing the STI, bypass test times, and AOTs for the analog channels, as well as the AOTs for the logic cabinets, master relays, and slave relays. A PRA 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 the TOP. This approach 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 between the analysis methods used by the TOP in the WCAP-10271 series of reports and those used in WCAP-14333 and WCAP-15376 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 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, as well as the test and maintenance activities performed on these systems, described 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 data have shown that a significant number of reactor trips are related to instrumentation test and maintenance activities, indicating that these activities should be completed with caution and that 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, Revision 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 TSs 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 STIs for the reactor trip breakers, master relays, logic cabinets, and analog channels.

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

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 CT 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, allows one Intermediate Range Neutron Flux channel to be inoperable for an indefinite period of time with power level above 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-0452, the justification for changing this action to Condition F from NUREG-1431 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. TSTF-246 was approved by the NRC staff by letter dated March 22, 1999 (ADAMS Legacy Library Accession No. 9903250187).

4.2 Summary of the TS Changes Proposed by the Licensee

The following table summarizes the proposed changes from WCAP-14333, as applicable to McGuire 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 changes from WCAP-15376, as applicable to McGuire 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[1]	2	6				
Analog Channels	3	6				
RTBs	2	4	1	24	2	4

1. Applicable to SSPS plants only.

4.3 Review of Methodology

In accordance with Sections 16.1, 19.1, and 19.2 of the SRP, 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 conditions and limitations from the SEs 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:

- The licensees application dated December 11, 2007, as supplemented December 18, 2008; especially Enclosure 1 and associated Attachments 5 and 6
- TSTF-411, Revision 1, and TSTF-418, Revision 2; as approved by SEs dated August 30, 2002, and April 2, 2003, respectively; and the
- The NRC staff SEs for WCAP-14333 and WCAP-15376.

The NRC staff also referred to SEs for WCAP-10271 and the licensee's individual plant examination (IPE) and individual plant examination of external events (IPEEE).

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 data have 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 from Section 19.2 of the SRP, 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 from Section 19.2 of the SRP, 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. 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 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 reports and used by the licensee to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177.

To determine whether WCAP-14333 and WCAP-15736 are applicable to McGuire 1 and 2, the licensee addressed the conditions and limitations of the NRC staff's SEs for these topical reports 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 McGuire 1 and 2.

The following paragraphs discuss the licensee's evaluation of the limitations and conditions found in the SEs for WCAP-14333 and WCAP-15376:

(1) A licensee should confirm the applicability of 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 with the containment not isolated. The NRC staff SE stated that 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.

McGuire 1 and 2 utilize an ice condenser containment. The licensee stated that the LERF for McGuire 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. This system 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 result 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 ratio. Both estimates are below RG 1.174 acceptance of 1.0E-7 for a very small change. Based on the qualitative and quantitative LERF evaluations, the licensee found that the plant-specific LERF contribution is consistent with the 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 McGuire 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 NRC staff concludes that the licensee addressed both Tier 2 and Tier 3 risk significant configurations and confirmed that these insights are incorporated into the McGuire 1 and 2 CRMP. Therefore, the NRC staff considers this condition satisfied for McGuire 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 McGuire 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 a response dated September 28, 2001 (ADAMS Accession No. ML012820263) to an NRC staff Request for Additional Information regarding WCAP-15376 issued August 1, 2001 (ADAMS Accession No. ML012140562) the PWROG provided generic risk estimates that assumed concurrent testing. The resulting ICCDP estimate was higher than the results from WCAP-15376, but within the acceptance guidelines of RG 1.177. Based on the applicability of WCAP-15376 to McGuire 1 and 2, and an ICCDP estimate that is 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 7A, table 5, of the licensee's submittal confirmed that the assumptions regarding human reliability used in WCAP-15376 are applicable to McGuire 1 and 2. The licensee's review concluded that for the operator actions identified in WCAP-15376 and credited in the McGuire PRA, plant procedures, training and sufficient time are available consistent with the assumptions in

WCAP-15376. Based on these considerations, 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. The McGuire 1 and 2 design is based on the SSPS, therefore, this condition is not applicable to the implementation of WCAP-15376 at McGuire 1 and 2.

(6) WCAP-15376 included an additional condition based on the PWROG commitment that each plant to 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. McGuire 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 McGuire 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 McGuire 1 and 2 TSs Sections 3.3.1, 3.3.2 to be within the scope of the NRC staff's SE on WCAP-15376 and, therefore, acceptable. Furthermore, the NRC staff finds that the licensee's regulatory commitments due date of within 90 days of NRC staff 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: PRA 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 McGuire 1 and 2. The Tier 1 NRC staff

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 provide a generic PRA model for the evaluation of the CT, test bypass time and STI extensions. Although the SEs for WCAP-14333 and WCAP-15376 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, as well as the NRC staff findings, limitations, and conditions contained in the SEs for WCAP-14333 and WCAP-15376. WCAP-14333 and WCAP-15376 do not require the use of the McGuire 1 and 2 PRA or plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP for the implementation of either topical report. However, in the SEs for WCAP-14333 and WCAP-15376, the NRC staff found that the generic PRA analysis for the proposed CT, bypass test time, and STI changes used in these topical reports may not be representative of other Westinghouse plants because of 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 McGuire 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, anticipated transient without scram (ATWS), total internal events CDF, transient events, operator actions, RTS trip actuation signals, and ESFAS actuation signals to plant specific values. Based on a comparison to the implementation guidelines and the conditions and limitations contained in the NRC staff's SEs for WCAP-14333 and WCAP-15376, the NRC staff concludes that WCAP-14333 and WCAP-15376 are applicable to McGuire 1 and 2.

Peer Review

The original IPE for McGuire 1 and 2 was submitted by letter dated November 4, 1991. Revision 2 was provided to the NRC staff in 1998. Revision 3 was completed in July 2002, and Revision 3a was completed February 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 McGuire 1 and 2 PRA August 18, 1987.

- An NRC SE was issued on the IPE and IPEEE for McGuire 1 and 2 by letters dated June 7, 1994, and April 12, 1999, respectively.

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 McGuire 1 and 2.

PRA Results and Insights

Cumulative Risk

WCAP-15376 evaluated the cumulative CDF risk from pre-TOP (prior to implementation of WCAP-10271) 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, a very small change, but within the acceptance guidelines for a small change. For McGuire 1 and 2, the cumulative risk is limited from the TOP condition (implementation of WCAP-10271) to implementation of WCAP-15376. Since the proposed change for McGuire 1 and 2 is from implementation of TOP to implementation of WCAP-15376, the NRC staff finds that 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 McGuire 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 SE 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 for the proposed 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 McGuire 1 and 2. The proposed changes will increase the unavailability of the affected SSCs 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 that 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 supplemental letter of December 18, 2008,

> The risk significant fire initiating events in the PRA are those fires that result in failure of the nuclear service water (RN) system. This was initially noted in the IPEEE report submitted to the NRC on June 21, 1994 (Section 4.8 and Appendix B

Section 3.5.4 of the IPEEE). The RN system provides cooling to the pumps for the majority of the mitigating systems. Consequently, when nuclear service water 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 (i.e., RN cools component cooling) leading to RCP seal LOCAs. In these sequences, core damage can be mitigated by 1) operator actions to start the train with no breaker control power from the switchgear room, 2) operator actions to swap RN from the other unit, or 3) manually starting 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.

External Fire +

Initiator Name*	Description	CDF for ESFAS/RTS Actuation Signal Failures in Fire Sequences	Fire Event CDF	Percent Contribution to Fire Event CDF
FVIC	Vital I&C Fire Causes A Loss Of RN	0/yr	6.1E-06/yr	0.0%
FCBLR	Cable Room Fire Causes A Loss Of RN	0/yr	1.2E-07/yr	0.0%
FASP	Aux. Shutdown Panel Fire Causes Loss Of RN	0/yr	7.7E-08/yr	0.0%
FCR	Control Room Fire Causes A Loss Of RN	0/yr	2.8E-08/yr	0.0%
FMFP	Main Feedwater Pump Fire	0/yr	2.8E-08/yr	0.0%
FACTB	All Consuming TB Fire Initiating Event	0/yr	2.3E-08/yr	0.0%
	Total Fire	0/yr	6.4E-06/yr	0.0%

*Other fire initiating events are included in the PRA model, but only the ones listed in this table 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. This review indicated that none of the fire sequences contain these types of failures. Therefore, the NRC staff found that 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 SE 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 a failure of secondary side heat removal along with a loss of power. A majority of the remaining sequences involved Reactor Coolant Pump seal LOCAs of varying sizes.
>
> ESFAS failures are explicitly modeled in the seismic PRA. The accident sequences from the CDF quantification for seismic events were reviewed to identify sequences that contain failures of the ESFAS.

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

Initiator Name	Description	Seismic + CDF for ESFAS Actuation Signal Failures in Seismic Sequences	Total Seismic CDF	Percent Contribution to Seismic CDF
SEISMIC	Seismic Initiator	3.0E-06/yr	8.9E-06/yr	33.7% #
SEISMIC	Seismic Initiator	4.5E-08/yr	1.1E-05/yr	0.4% *

+Applies to both Units 1 and 2.
Current model
* IPEEE model

The ESFAS failures are not a dominant contributor to the seismic PRA results when the IPEEE model is used. 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. The licensee concluded that the current seismic model percent contribution to the McGuire 1 and 2 seismic CDF due to ESFAS actuation signal failures (i.e. 34 percent) is conservatively elevated due to the use of plant level fragilities. This percentage is expected to drop to less than 1 percent when component level fragilities are reinstated. 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 McGuire 1 and 2 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 provided 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	1.6E-06/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 McGuire 1 and 2 IPE and IPEEE concluded that the contribution to plant risk from external flooding, transportation, and nearby facility accidents is not significant.

Therefore, the NRC staff finds that, 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

McGuire 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 2.2E-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 licensee 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 Auxiliary Feedwater (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., Alternating Current (AC) and Direct Current (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 operable for maintenance.

For WCAP-15376:

- The probability of failing to trip the reactor on demand will increase when a RTB train is removed from service; therefore, systems designed for mitigating an ATWS event should be maintained and available. RCS 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) 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 McGuire 1 and 2. Based on the above, the NRC staff finds that the licensee's Tier 2 analysis supports the implementation of WCAP-14333 and WCAP-15376 at McGuire 1 and 2 and satisfies the condition regarding Tier 2 from the NRC staff SEs for WCAP-14333 and WCAP-15376.

4.6.4 Tier 3: Risk-Informed CRMP

Risk assessment of online configurations for McGuire 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 McGuire 1 and 2 risk assessment guidelines use the results of the McGuire 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, then 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 that McGuire 1 and 2 satisfies 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 SEs. 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 SE. As such, the NRC staff finds that the implementation of WCAP-14333 and WCAP-15376 at McGuire 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. The changes these changes will be evaluated in a future amendment.

4.9 NRC Staff Findings and Conditions

The NRC staff finds that the licensee has demonstrated the applicability of WCAP-14333 and WCAP-15376 to McGuire 1 and 2 and has met the limitations and conditions as outlined in the NRC staff SEs. 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 McGuire 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 SE limitations and conditions and 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 that 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 SE conditions for these topical reports.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North 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, "Standards for Protection Against Radiation," 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 15781). 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.

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Date: December 30, 2008

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,

/RA/

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

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 248 to NPF-9
2. Amendment No. 228 to NPF-17
3. Safety Evaluation

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