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

September 12, 2011

Mr. Regis T. Repko
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 TECHNICAL SPECIFICATION CHANGES TO ALLOW MANUAL OPERATION OF THE CONTAINMENT SPRAY SYSTEM (TAC NOS. ME4051 AND ME4052)

Dear Mr. Repko:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 265 to Renewed Facility Operating License NPF-9 and Amendment No. 245 to Renewed Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 28, 2010, as supplemented by letters dated November 15, 2010, March 23, 2011, and May 2, 2011.

The amendments revise the TSs to allow manual operation of the containment spray system and to change the setpoints for the refueling water storage tank.

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 301-415-1119.

Sincerely,

Jon Thompson

Jon Thompson, 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. 265 to NPF-9
2. Amendment No. 245 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. 265
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 May 28, 2010, as supplemented by letters dated November 15, 2010, March 23, 2011, and May 2, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of 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. 265 , are hereby incorporated into this renewed operating 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 prior to the first entry into Mode 4 after the refueling outage where all of the modifications associated with the amendment have been completed.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, 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: September 12, 2011

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. 245
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 May 28, 2010, as supplemented by letters dated November 15, 2010, March 23, 2011, and May 2, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of 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. 245 , are hereby incorporated into this renewed operating 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 prior to the first entry into Mode 4 after the refueling outage where all of the modifications associated with the amendment have been completed.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, 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: September 12, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 265

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 245

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
TS Pages	TS Pages
3.3.2-10	3.3.2-10
3.3.2-11	3.3.2-11
3.3.2-12	3.3.2-12
3.3.2-13	3.3.2-13
3.3.2-14	3.3.2-14
3.3.2-15	3.3.2-15
3.5.4-2	3.5.4-2
3.6.6-1	3.6.6-1
3.6.6-2	3.6.6-2

(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. 265 , 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. 245, 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.

Table 3.3.2-1 (page 1 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≤ 1.2 psig	1.1 psig
d. Pressurizer Pressure - Low Low	1,2,3(a)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 1835 psig	1845 psig
2. Containment Spray*						
a. Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≤ 3.0 psig	2.9 psig
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.7	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA

(continued)

* The requirements of this function are not applicable following implementation of the modifications associated with ECCS Water Management on the respective Unit.

(a) Above the P-11 (Pressurizer Pressure) interlock.

Table 3.3.2-1 (page 2 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation (continued)						
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.7	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8	≤ 3.0 psig	2.9 psig
4. Steam Line Isolation						
a. Manual Initiation						
(1) System	$1,2^{(b)},3^{(b)}$	2 trains	F	SR 3.3.2.7	NA	NA
(2) Individual	$1,2^{(b)},3^{(b)}$	1 per line	G	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	$1,2^{(b)},3^{(b)}$	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	$1,2^{(b)}, 3^{(b)}$	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≤ 3.0 psig	2.9 psig
d. Steam Line Pressure						
(1) Low	$1,2^{(b)}, 3^{(a)(b)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 755 psig	775 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
(b) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 3 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	$3^{(b)(c)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	$\leq 120^{(d)}$ psi	$100^{(d)}$ psi
5. Turbine Trip and Feedwater Isolation						
a. Turbine Trip						
(1) Automatic Actuation Logic and Actuation Relays	1,2	2 trains	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(2) SG Water Level-High High (P-14)	1,2	3 per SG	J	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.8 SR 3.3.2.9	≤ 85.6%	83.9%
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements. See item 5.a.(1) for Applicable MODES.					
b. Feedwater Isolation						
(1) Automatic Actuation Logic and Actuation Relays	$1,2^{(e)}, 3^{(e)}$	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(2) SG Water Level-High High (P-14)	$1,2^{(e)}, 3^{(e)}$	3 per SG	D	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.8 SR 3.3.2.9	≤ 85.6	83.9%

(continued)

(b) Except when all MSIVs are closed and de-activated.
(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Steam Line Isolation Steam Line Pressure-Low is not blocked.
(d) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5. Turbine Trip and Feedwater Isolation (continued)						
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements. See Item 5.b.(1) for Applicable MODES.					
(4) Tavg-Low	1,2(e)	1 per loop	J	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8	≥ 551°F	553°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
(5) Doghouse Water Level-High High	1,2(e)	2 per train per Doghouse	L,M	SR 3.3.2.1 SR 3.3.2.7	≤ 13 inches	12 inches
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - Low Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 15%	16.7%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Station Blackout						
(1) Loss of voltage	1,2,3	3 per bus	D	SR 3.3.2.7 SR 3.3.2.9	≥ 3122 V (Unit 1) ≥ 3108 V (Unit 2) with 8.5 ± 0.5 sec time delay	3174 V (Unit 1) 3157 V (Unit 2) ± 45 V with 8.5 ± 0.5 sec time delay
(2) Degraded Voltage	1,2,3	3 per bus	D	SR 3.3.2.7 SR 3.3.2.9	≥ 3661 V (Unit 1) ≥ 3685.5 V (Unit 2) with ≤ 11 sec with SI and ≤ 600 sec without SI time delay	3678.5 V (Unit 1) 3703 V (Unit 2) with ≤ 11 sec with SI and ≤ 600 sec without SI time delay

(continued)

(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

Table 3.3.2-1 (page 5 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
e. Trip of all Main Feedwater Pumps	1,2	1 per MFW pump	K	SR 3.3.2.7 SR 3.3.2.9	NA	NA
f. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	2 per MDP, 4 per TDP	N,O	SR 3.3.2.7 SR 3.3.2.8 SR 3.3.2.9	≥ 3 psig	3.5 psig
7. Automatic Switchover to Containment Sump						
a. Refueling Water Storage Tank (RWST) Level - Low	1,2,3	3	P,S	SR 3.3.2.1 SR 3.3.2.3(a)(b) SR 3.3.2.8(a)(b) SR 3.3.2.9	≥ 175.85 inches*	180 inches*
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

* Following implementation of the modifications associated with ECCS Water Management on the respective Unit, the Allowable Value for this Function shall be ≥ 92.3 inches and the Nominal Trip Setpoint for this Function shall be 95 inches.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the UFSAR.

Table 3.3.2-1 (page 6 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.7	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	Q	SR 3.3.2.5 SR 3.3.2.8	$\leq$ 1965 psig	1955 psig
c. T_{avg} - Low Low, P-12	1,2,3	1 per loop	Q	SR 3.3.2.5 SR 3.3.2.8	$\geq$ 551°F	553°F
9. Containment Pressure Control System	1,2,3,4	4 per train, 2 trains	R	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.8	Refer to Note 1 on Page 3.3.2-14	Refer to Note 1 on page 3.3.2-14

NOTE 1: The Trip Setpoint for the Containment Pressure Control System start permissive/termination (SP/T) shall be $\geq$ 0.3 psig and $\leq$ 0.4 psig. The allowable value for the SP/T shall be $\geq$ 0.25 psig and $\leq$ 0.45 psig.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is ≥ 70°F and ≤ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 372,100 gallons.*	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

* Following implementation of the modifications associated with ECCS Water Management on the respective Unit, the RWST borated water volume for this SR shall be ≥ 383,146 gallons.

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1*	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

* Following implementation of the modifications associated with ECCS Water Management on the respective Unit, there will be no automatic valves in the Containment Spray System.

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3*	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.4*	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.5**	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6***	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage

* Following implementation of the modifications associated with ECCS Water Management on the respective Unit, the requirements of SR 3.6.6.3 and SR 3.6.6.4 shall no longer be applicable.

** Following implementation of the modifications associated with ECCS Water Management on the respective Unit, SR 3.6.6.5 is revised to state the following: Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to manually start upon receipt of a start permissive from the Containment Pressure Control System (CPCS)

*** Following implementation of the modifications associated with ECCS Water Management on the respective Unit, SR 3.6.6.6 is revised to state the following: Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to manually open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 265 TO RENEWED FACILITY OPERATING LICENSE NPF-9

AND

AMENDMENT NO. 245 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 May 28, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101600256), as supplemented by letters dated November 15, 2010 (ADAMS Accession No. ML103270051), March 23, 2011 (ADAMS Accession No. ML110840443), and May 2, 2011 (ADAMS Accession No. ML11124A125), Duke Energy Carolinas, LLC (Duke Energy, the licensee), submitted a proposed license amendment to change the McGuire 1 and 2 Technical Specifications (TSs). The supplements dated November 15, 2010, March 23, 2011, and May 2, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published the *Federal Register* on (75 FR 61524).

The proposed change would revise the TSs to allow manual operation of the containment spray system (CS) and to change the setpoints for the refueling water storage tank (RWST).

2.0 REGULATORY EVALUATION

The regulatory requirements which the NRC staff considered in its review of the license amendment request (LAR) are as follows:

- the regulation at Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36(c)(1)(ii)(A) states in part

  Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting

> safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor....

- the regulation at 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"
- the regulation at 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"
- the regulation at 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants"
- the regulation at 10 CFR 50.67, "Accident source term"
- the regulation at 10 CFR 50.120, "Training and qualification of nuclear power plant personnel"
- the regulations at 10 CFR, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 13 (GDC 13), "Instrumentation and control"
- GDC 16, "Containment design"
- GDC 17, "Electric power systems"
- GDC 18, "Inspection and testing of electric power systems"
- GDC 19, "Control room"
- GDC 20, "Protection system functions"
- GDC 21, "Protection System Reliability and Testability"
- GDC 35, "Emergency core cooling "
- GDC 38, "Containment heat removal"
- GDC 39, "Inspection of containment heat removal system"
- GDC 40, "Testing of containment heat removal system"
- GDC 41, "Containment atmosphere cleanup"
- GDC 42, "Inspection of containment atmosphere cleanup systems"
- GDC 43, "Testing of containment atmosphere cleanup systems"
- GDC 50, "Containment design basis"

- the regulation at 10 CFR, Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities"

- the regulation at 10 CFR Part 55, "Operators' Licenses"

The regulatory guidance and other documentation which the NRC staff also considered in its review of the LAR are as follows:

- Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps"

- RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and remain within, the TS limits

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

  The proposed change impacts the post-accident sump pH profile. The NRC staff reviewed the LAR with respect to the licensee's analysis for maintaining sump pool pH greater than or equal to 7 for 30 days following a loss-of-coolant accident (LOCA). According to RG 1.183, maintaining pH basic will minimize re-evolution of iodine from the sump pool water.

- NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2

- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 2

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

  Chapter 6, "Engineered Safety Features," addresses containment analyses in several of its sections. In particular, Sections 6.2.1, "Containment Functional Design," 6.2.1.1.B, "Ice Condenser Containments," 6.2.2," Containment Heat Removal Systems," and 6.4, "Control Room Habitability System," were considered in the technical evaluation.

  Chapter 13, "Conduct of Operation," addresses several human factors engineering topics. Specific sections considered in this review were sections 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," and 13.5.2.1, "Operating and Emergency Operating Procedures."

  Chapter 15, "Accident Analysis," provides regulatory guidance for analysis of design-basis accidents (DBAs).

  Chapter 18, "Human Factors Engineering" provides review guidance for this topic.

- NUREG-1431, Revision 3, "Standard Technical Specifications - Westinghouse Plants"

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plant"

- NUREG-1764,"Guidance for the Review of Changes to Human Actions"
- NUREG/CR-5950, "Iodine Evolution and pH Control"
- Regulatory Issue Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006, describes an acceptable method to meet the requirements of 10 CFR 50.36, "Technical specifications"
- Generic Letter (GL) 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability"
- GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"
- Generic Safety Issue 191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance"
- Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times"
- NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors"

3.0 TECHNICAL EVALUATION

3.1 Description of Proposed Change

The purpose of this amendment request is to increase the available water in the RWST for the emergency core cooling system (ECCS), to reduce the probability of transfer to containment sump recirculation, and to increase operator response time before the transfer to containment sump recirculation. In addition, implementation of the proposed changes may reduce the debris loading on the containment sump strainer assemblies as recommended in NRC Bulletin 2003-01.

The licensee has proposed to lower the allowable value (AV) and nominal trip setpoint (NTSP) for the RWST Level-Low function in TS Table 3.3.2-1, Function 7a. This change is based on the reduced tank depletion rate following the removal of the automatic CS operation and changes in the vortexing allowance based on testing and analytical refinements. This change will also incorporate the footnotes contained in Technical Specification Task Force Change Traveler (TSTF)-493, Revision 4, for TS Table 3.3.2-1, Function 7a.

The licensee also proposed to increase the RWST minimum volume required by TS 3.5.4 from 372,100 gallons (equivalent to 456 inches of the indication level) to 383,146 gallons (equivalent to 470 inches of the indication level), resulting in a usable volume increase of 11,046 gallons. The RWST ensures that an adequate supply of borated water is available to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and CS pump operation in the recirculation mode.

The licensee will make the following hardware modifications with the above proposed TS changes:

- Adjust the RWST low level setpoint (for automatic switchover to the containment sump).
- Adjust the RWST low level and low-low-level alarm setpoints.
- Install new redundant wide-range RWST pre-low-level annunciator alarms.
- Modify the existing narrow range (NR) RWST level instrumentation to improve accuracy and support a higher TS minimum limit.

In addition to those hardware modifications, a combination of system alignment and procedural guidance will also preclude CS pump operation. Procedural guidance will allow one CS pump to be operated manually only when aligned to the containment sump. This will minimize water draindown from the RWST and allow a longer response time to transfer the ECCS pump suction to the containment sump, thus allowing a reduction in the RWST low- and low-low-level setpoints. In addition, the normal CS alignment is such that no single failure will result in the depletion of RWST inventory by CS pump operation. Therefore, the RWST low-level setpoints may be reduced accordingly.

3.2 Instrumentation and Control

3.2.1 TS Table 3.3.2-1, Function 7a, Automatic Switchover to Containment Sump, RWST Level-Low

The licensee revised the AV and NTSP from greater than or equal to 175.85 inches and 180 inches to greater than or equal to 92.3 inches and 95 inches, respectively.

The licensee provided the basic setpoint calculation methodology and the summary of setpoint calculations for the proposed setpoint changes. For the RWST low level, the licensee established the analytical limit of 76.9 inches and then calculated the total loop uncertainty as ±10 inches. Based on the analytical limit and total loop uncertainty values, the licensee established the NTSP of 95 inches, which includes a margin of 8.1 inches. The licensee calculated the AV based on the uncertainty associated with the portion of the loop tested for the desired interval and also on the uncertainty of the loop not being tested. The more conservatively calculated value of these two methods was then used as the AV. As a result, the licensee established the AV for low-level RWST as 92.3 inches. The licensee calculated as-found tolerance (AFT) based on the uncertainties, which include reference accuracy, drift, and measurement and test equipment uncertainties. The licensee calculated as-left tolerance (ALT) based on the uncertainties, which include reference accuracy, measurement and test equipment, and reading resolution. The licensee also stated that the previous calibration or surveillance as-left setting value for a channel will be used as the starting point for determining if the next surveillance AFT is met. The calculated values of AFT and ALT were determined to be ±1.67 inches and ±0.5 inches, respectively. The method used to calculate these terms meets NRC staff guidance in RIS 2006-17, RG 1.105, Revision 3, and meets the requirements of 10 CFR 50.36(c)(1)(ii)(A).

The licensee performed setpoint calculations in accordance with the Duke Energy Engineering Directives Manual (EDM)-102, "Instrument Setpoint/Uncertainty Calculations." The methodology described in EDM-102 is consistent with the intent of Instrument Society of America Standard RP67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation."

In order to define the operability of the instrument loop in the TS, the licensee added two footnotes to Function 7a in TS Table 3.3.2-1. These two footnotes are consistent with TSTF-493, Revision 4, and meet the NRC guidance in RIS 2006-017. These two footnotes enhance safety by ensuring that unexpected as-found conditions are evaluated before returning the channel to service and by ensuring that as-left settings provide sufficient margin for uncertainties.

Based on the above discussion, the NRC staff finds that the proposed TS change meets the regulatory requirements found in 10 CFR 50.36(c)(1)(ii)(A), is consistent with NRC guidance found in RG 1.105, Revision 3, and RIS 2006-17, and is consistent with the NRC staff-approved TSTF-493, Revision 4.

3.2.2 TS Surveillance Requirement (SR) 3.5.4.2 Verify Reserve Water Storage Tank Borated Water Volume

The licensee proposed to revise TS SR 3.5.4.2 to increase the minimum water volume in the RWST from greater than or equal to 372,100 gallons (equivalent to 456 inches) to greater than or equal to 383,146 gallons (equivalent to 470 inches). TS SR 3.5.4.2 ensures that the minimum volume of borated water in the RWST is available to the ECCS and CS for accident mitigation.

The licensee will replace the existing NR RWST level transmitters to reduce the overall instrument loop measurement uncertainty and allow a higher minimum RWST water volume to be maintained. One non-safety-related NR instrument loop is provided for each of the two RWSTs at McGuire 1 and 2. This NR RWST instrument loop is used to verify the pre-accident initial conditions of the RWST volume in accordance with TS SR 3.5.4.2. These instruments are only used for the surveillance and do not actuate automatic or manual operator actions for accident mitigation.

The NRC staff had concerns about how the accuracy of these instruments is maintained and the operability requirements of the instruments. In response to the NRC staff's request for additional information (RAI), the licensee provided the calculation of the instrumentation loop uncertainty analysis by letter dated November 15, 2010, including required calibration tolerances associated with the loop readout devices. The following are the required loop calibration tolerances with the span of NR instrumentation from 450 to 500 inches for the three readout devices:

- NR alarm device setting tolerances: less than or equal to ±0.50 percent span (equivalent to ±0.25 inch)
- NR main control board (MCB) indication device setting tolerances: less than or equal to ±2.0 percent span (±1.0 inch)
- NR operational aid computer (OAC) device setting tolerances: less than or equal to ±0.30 percent span (±0.15 inch)

The licensee stated that the OAC display is the operator's preferred indication to verify the TS required minimum volume in the RWST; however, the main control room board indication can also be used. The TS surveillance criterion for the minimum RWST volume of 383,146 gallons will be greater than or equal to 471 inches with the OAC display and 472 inches with the main control room board indication. The 7-day surveillance frequency is based on the current frequency specified in the SR bases in NUREG-1431 and has been shown to be acceptable through operating experience. The modified RWST NR level loop will provide computer and annunciator alarms with early warning of low water level before the RWST level falls below the TS-required minimum level.

Based on the fact that these instruments are used for the fulfillment of the SR with sufficient accuracies and that the surveillance frequency is consistent with Standard TS SR bases found in NUREG-1431, the NRC staff finds that the proposed TS change is acceptable.

3.3 Containment

The licensee requested changes to the McGuire 1 and 2 TS to modify the CS actuation logic by changing the automatic start of the CS to a manual start during a DBA. A revised containment analysis is performed in order to demonstrate containment integrity with the proposed change. The analysis evaluated peak containment pressure, maximum sump liquid temperature, and maximum containment vapor temperature for various break scenarios. The peak containment pressure and maximum sump liquid temperature is obtained as a result of a large-break LOCA located on a cold leg pump discharge pipe. The maximum containment vapor temperature resulted from a large steam line break (SLB).

3.3.1 Large-Break LOCA Containment Analysis

For the large-break LOCA containment analysis, the licensee used the NRC-approved ice condenser containment analysis methodology described in Reference 1, modified per Attachment 4 of Reference 2, by including the deletion of automatic spray actuation logic and replacing it with its manual actuation. In Reference 3, the NRC staff approved the requested modifications. The methodology uses the GOTHIC computer code for the containment analysis and the RELAP5 computer code for the mass and energy release calculation. The NRC staff finds this acceptable as the containment analysis methodology being used has already been reviewed and approved in a prior NRC staff safety evaluation.

The proposed containment analysis includes changes from the current licensing basis (CLB) analysis which are (a) higher initial water inventory in the RWST, (b) a lower value for the RWST low level alarm setpoint, (c) a lower value for the RWST low-low alarm setpoint, (d) deletion of the automatic initiation of the CS.

The analysis also reflected changes in operator actions resulting from the proposed revisions in the emergency procedures. A description of these are as follows: (a) transfer of the suction of the high head and intermediate head safety injection pumps to the residual heat removal (RHR) pump discharge is done upon receiving the RWST low-low level alarm signal, and (b) manual starting of one containment spray pump after verification of adequate sump level, and if containment pressure is greater than 3 pounds per square inch gauge (psig), after the suction transfer of the RHR pump to the containment sump at the RWST low level alarm setpoint. The licensee states:

> The timing associated with the manual start of one CS Pump by the operators is dependent upon the single failure assumed. For most cases, the CS Pump would be

aligned prior to reaching the RWST low-low level. When the single failure affects the valves that automatically swap during the RHR transfer to the containment sump, the CS Pump may be aligned after reaching the RWST low-low level.

The changes in operator actions resulting from the proposed revisions in the emergency procedures also include: (c) not to use auxiliary containment spray.

The limiting peak containment pressure resulted from the assumption of a cold leg pump discharge pipe break LOCA and a single failure which minimized the spray flow and the heat exchangers available for containment heat removal. The analysis gave a peak containment pressure of 13.87 psig compared to the current Updated Facility Safety Analysis Report (UFSAR) peak pressure of 13.07 psig. The current UFSAR containment design pressure is 15.0 psig. The increase in peak pressure is due to the decrease in the integrated containment spray flow over the transient period. The NRC staff considers this analysis acceptable because the resulting limiting peak containment pressure is less than the current UFSAR containment design pressure.

The limiting sump liquid temperature resulted from the assumption of a hot leg break LOCA and a single failure of one engineered safety feature (ESF) train which minimized the spray flow and the heat exchangers available for containment heat removal. The licensee analyzed the hot leg break LOCA for maximum and minimum ECCS flows and found that the case with the maximum ECCS flow resulted in maximum sump liquid temperature. The maximum sump liquid temperature was calculated to be 197 °F which occurs at the time of suction transfer to the sump. This temperature is greater than the current UFSAR peak value of 190 °F for approximately 9 minutes. The higher temperature is due to absence of containment spray flow mixing with the sump fluid until the time of suction transfer to the sump. The NRC staff considers the analysis acceptable because the licensee's analyses still showed acceptable results despite this increase in maximum sump liquid temperature.

The licensee performed a sensitivity study to evaluate the containment pressure response to a potential failure of an operating CS pump. In the analysis, the operating pump is assumed to fail at the time of peak pressure during the transient which is the limiting scenario. The result of the study showed that if the spray flow is reinitiated up to 20 minutes from the time of failure of the operating pump, the peak containment pressure will remain below the containment design pressure of 15 psig. The NRC staff considers the analysis acceptable, because 20 minutes is a reasonably conservative time for an operator to start the standby pump in order to reinitiate the spray flow and the peak containment pressure will remain below the containment design pressure for this period of time.

3.3.2 Large Steam Line Break Containment Analysis

The current large SLB analysis predicts that the average lower containment vapor temperature peaks within the first 30 seconds and drops to between 240 °F and 250 °F by 60 seconds. Due to the timing of spray initiation, the containment sprays do not affect the lower containment temperature. Also the analysis did not include the actuation of the containment air return fans because they would not automatically start during the time of interest for this event. Therefore, the licensee concludes that the current SLB analysis bounds the containment response with the proposed modification which eliminates the automatic actuation of the CS. The NRC staff has reviewed the licensee's evaluation and has concluded that it is acceptable based on the fact that the current SLB analysis bounds the containment response described in the LAR. This is because, during the time period of interest, spray initiation does not affect containment response and the containment air return fans would not automatically start and, therefore, would not affect the containment response.

3.3.3 ECCS Pumps Net Positive Suction Head (NPSH) Evaluation

The licensee stated that the available NPSH for the current ECCS pumps does not take credit for the containment pressure above the atmospheric pressure or the static head of the sump fluid above the elevation of the containment sump floor during a DBA. For the proposed NPSH analysis, consistent with the guidance of RG 1.1, the licensee did not credit the post-accident containment pressure, but credited 3 feet of sump liquid level which is the minimum level required for strainer submergence. The licensee calculated the available NPSH at the maximum sump water temperature and assumed maximum head loss for the strainer debris loading. Under these conditions, the proposed analysis resulted in more than 2 feet of margin in the limiting value of available NPSH above its required value. In a letter dated March 23, 2011, the license responded to RAI questions including those regarding (a) the basis for required NPSH that was used to compare it with the available NPSH and (b) a list of the uncertainties included in evaluation of the required NPSH. In its response, the licensee stated that:

> Based on the aggregate conservatisms outlined in response to items 4(A) and (C) here-in, no explicit additional uncertainties were accounted for within the NPSH calculation.

(Item 4(A) describes the basis of the required NPSH that was used to compare it with the available NPSH and Item 4(C) outlines all the conservatisms used in the calculation of the limiting value of available NPSH).

Based on the information provided by the licensee, the NRC staff considers that the changes to the NPSH evaluation resulting from the LAR are acceptable. The NRC staff notes, in particular, that the assumption of zero psig in the containment and the conservative assumptions about sump level for the calculation of available NPSH provide adequate margin and preclude the need for explicit uncertainty in the required NPSH. For this reason, as well as the limited duration of the increased sump fluid temperature, the NRC staff concludes that the effect of implementing the LAR would either be neutral or salutary with respect to the performance of the suction strainer and the NPSH evaluation.

3.3.4 Minimum Containment Pressure Analysis for ECCS Evaluation

The licensee stated in the LAR that:

> The proposed change to CS will not adversely impact the minimum containment pressure analysis included in the peak clad temperature analysis. The absence of CS would be expected to increase the minimum containment pressure as a function of time. However, for ice condenser plants, the increase in containment pressure resulting from the elimination of CS would be limited.

The NRC staff finds the licensee's evaluation of the minimum containment pressure acceptable because the analysis in Section 3.3.1 of this safety evaluation shows that the containment pressure is higher in the absence of sprays and, therefore, the minimum containment pressure would not be adversely impacted.

3.3.5 Anticipated Transient without Scram Event

During an anticipated transient without scram (ATWS) event, the pressurizer relief valve lift will cause blowdown of the pressurizer into the pressurizer relief tank eventually causing the pressurizer relief tank rupture disks to break releasing steam inside the containment. The licensee stated:

> The associated mass and energy release due to the blowdown of the pressurizer relief tank will not produce a limiting containment pressure response.

The licensee also stated:

> If the containment pressure were to increase to the high-high containment pressure setpoint, the operation of the ARS [containment air return] will be sufficient to ensure an acceptable containment pressure response.

The NRC staff finds the licensee's evaluation of the ATWS event acceptable because even if the pressurizer relief tank rupture disks were to break and release steam inside the containment and cause containment pressure to increase to the high-high containment pressure setpoint, the analyses show that the operation of the ARS will be sufficient to ensure an acceptable containment pressure response.

3.3.6 Ice Condenser Lower Inlet Door Evaluation

The ice condenser lower inlet doors are required to open during a DBA to allow the flow of hot steam and air mixture into the ice condenser for removing its heat and thereby limit the containment peak pressure and temperature. The doors are designed to fully open at a one pound per square foot differential (psfd) pressure. During a small-break accident, the design requirement for each ice condenser door is to partially open by approximately the same amount.

In the event of a DBA, the lower inlet doors of the ice condenser open due to the pressure rise in the lower containment allowing air and steam to enter into the ice condenser. The pressure increase in the ice condenser causes the intermediate and top deck doors to open allowing the air to enter the containment upper compartment. The licensee states that:

> The lower inlet doors are designed to open in response to a 1 psf [pounds per square foot] pressure differential. An automatic CSS [Containment Spray System] actuation would occur at a containment pressure of 3 psig. This would correspond to a 432 psf pressure differential across the lower inlet doors, assuming the doors did not open. Therefore, it is expected that absent a CSS actuation, the lower inlet doors will respond as designed in the same manner as they would with the automatic signal in place. As a result, there will be no impact to the lower inlet door surveillance requirements.

The NRC staff has reviewed the licensee's evaluation and has concluded that the opening of the lower inlet doors of the ice condenser in response to small-break LOCAs is determined by the local lower compartment pressure. The lower inlet doors are designed to open at a one psfd pressure. In the current TSs, the automatic containment spray initiation occurs at three psig (432 pounds per square foot gauge) pressure across the doors. Therefore, in the event of a small-break accident, the absence of CS actuation will not impact the performance of the doors as the lower inlet doors are designed to open at a much lower pressure. The NRC staff has reviewed the licensee's evaluation and has concluded that it is acceptable based on the fact that during small-break LOCAs the differential pressure that results in opening of the doors is not reduced in the absence of CS actuation.

3.3.7 Early Containment Air Return Fan Operation

The licensee stated:

> The analyses that supported the early ARS fan operation submittal considered small breaks that did not reach the CSS actuation setpoint. Therefore, the proposed modification to remove the CSS actuation logic will not impact the analyses that support the associated Safety Evaluation Report.

The NRC staff has reviewed the licensee's evaluation and has concluded that the proposed LAR modification to remove the CS actuation logic will not impact the CLB as the analyses that supported the early ARS fan operation considered small breaks that did not reach the CS actuation setpoint.

3.3.8 Minimum Containment Sump Level Evaluation

The licensee stated that:

> The proposed modifications will increase the available RWST liquid between the TS minimum and the RWST low level and RWST low-low level alarms. The proposed modifications will also eliminate the automatic CSS actuation logic, eliminating any upper containment holdup penalty prior to reaching sump recirculation. These modifications will result in additional liquid inventory in the containment sump, and thus a higher sump level. Therefore, the proposed modifications will not adversely impact the minimum containment sump level analysis.

The NRC staff has reviewed the licensee's evaluation and has concluded that the proposed LAR modifications will not adversely impact the minimum containment sump level analysis as these modifications will result in a higher sump level.

3.3.9 Post-Accident Containment Sump pH

After a LOCA, a variety of different chemical species are released from the damaged core. One of them is radioactive iodine. This iodine, when released to the outside environment, will significantly contribute to radiation doses. It is, therefore, essential to keep it confined within the plant's containment. According to NUREG-1465, iodine is released from the core in three different chemical forms; at least 95 percent is released in ionic form as cesium iodide (CsI) and the remaining 5 percent as elemental iodine (I_2) and hydriodic acid (HI); the release contains at least 1 percent of each I_2 and HI.

CsI and HI are ionized in water and are, therefore, soluble. However, elemental iodine is scarcely soluble. To sequester the iodine in water, it is desirable to maintain as much as possible of the released iodine in ionic form. In the radiation environments existing in containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At a higher pH, conversion to elemental form is lower and at pH greater than 7 it becomes negligibly small. The relationship between the rate of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950.

In order to show that the proposed change in CS actuation time and RWST low level set point would not adversely impact dose calculations associated with iodine evolution, the licensee provided the results of a revised sump pH calculation.

The licensee provided information regarding the assumptions and calculations used to verify that the sump pH would remain greater than 7 following a LOCA. A detailed calculation was previously reviewed and approved by the NRC staff as part of an alternate source term (AST) amendment for McGuire 1 and 2 (Reference 4). The licensee's previous analysis considered minimum and maximum boron concentrations and volumes for the RWST, accumulators, pressurizer, and reactor coolant system (RCS). Additional inputs included the impact of strong acids generated by radiation of cable insulation and sump water. The revised calculation for this proposed amendment used modified values for RWST volume, ice melt time, system flow rates, and temperature profile. All other inputs to the calculation remained the same as the licensee's CLB which was previously approved by the NRC staff as part of the AST amendment. The revised calculation showed a slightly higher minimum sump pH and a slightly lower equilibrium sump pH when compared to the CLB. The differences in both cases were a tenth of a pH unit. Despite the minor pH differences between the calculations, the pH at reference temperature remained greater than 7 at all times in the revised calculation.

The NRC staff reviewed the licensee's assumptions and analyses and concluded that conservative values were used for the key parameters of the calculation. The assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment sump pH. The NRC staff finds the proposed changes acceptable with respect to their impact on post-LOCA containment sump pH.

3.3.10 Impact on Resolution of GL 2004-02

The proposed LAR could have an impact on resolution of GL 2004-02, but the impacts are not expected to be significant. Although the pH remains greater than 7 under the proposed conditions, it changes slightly from the CLB. Specifically, the minimum pH is slightly higher and equilibrium pH is slightly lower. As stated previously, the differences between the proposed change and the CLB are on the order of a tenth of a pH unit. This small change is not expected to have a significant impact. However, the licensee's chemical effects analysis, which is part of the resolution of GL 2004-02, will be slightly impacted by the change in the post-LOCA pH profile. Specifically, an increase in minimum pH may increase the rate of aluminum corrosion. In addition the slightly lower equilibrium pH may decrease the solubility of aluminum, resulting in increased chemical precipitation. The increase in maximum sump pool temperature will also impact the analysis of aluminum corrosion and precipitation. The NRC staff currently has open items regarding McGuire 1 and 2's chemical effects head loss testing for resolution of GL 2004-02 (Reference 5). A final determination regarding suction strainer performance, including resolution of the remaining RAI questions and verification that the final head loss test results are acceptable in light of the chemical effects analysis resulting from this LAR, will be made at the conclusion of that review.

3.3.11 Change in TS Table 3.3.2-1

The LAR proposes to delete functions 2a, 2b, and 2c from TS Table 3.3.2-1 following implementation of the proposed modifications. The NRC staff did not have any RAI questions regarding deletion of functions 2b and 2c because these are related with automatic actuation of the CS system, but it did have an RAI question regarding deletion of function 2a, which is related to the manual actuation of the CS system.

In a letter dated May 2, 2011, the licensee responded to the RAI question requesting the justification for deletion of function 2a by stating that:

> ... ESFAS [engineered safety feature actuation system] functions are performed by the Westinghouse Solid State Protection System (SSPS) using bistable and switch inputs from the W7300 process control system, field instruments and Main Control Board (MCB) manual switches.

The licensee further stated that:

> Following implementation of the associated modifications, circuitry controlling the containment spray pumps and related motor valves will no longer go through the SSPS. Actuation and interlock circuitry will remain safety-related, but will be implemented instead in auxiliary relay cabinets....
>
> Manual containment spray pump operation will continue to be tested periodically as required by SR 3.6.6.2... In addition, the manual start/stop push-buttons must be used in order to test the Containment Pressure Control System (CPCS), which is verified at 18 month intervals by SR 3.6.6.5.

The NRC staff concludes that deletion of 2a, 2b, and 2c from TS Table 3.3.2-1 following implementation of the proposed modifications is acceptable as functions 2b and 2c will not be necessary upon the removal of automatic CS actuation and manual containment spray pump operation (function 2a) will continue to have safety-related circuitry and will be addressed by SR 3.6.6.2 and SR 3.6.6.5.

3.3.12 Summary of Containment Performance Evaluation

The NRC staff has concluded, based on the evaluation of the proposed changes discussed above, that (1) the containment and associated systems will maintain a leak-tight barrier against the uncontrolled release of radioactivity to the environment and the containment design conditions important to safety are not exceeded, (2) the containment heat removal system will continue to perform its safety function in response to the DBAs, and (3) the containment heat removal system will maintain the containment structure and its internal compartments without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any DBA.

Although, as noted above, the NRC staff expects that approval of this LAR will have a neutral or salutary effect on sump strainer performance, the changes resulting from this LAR pertinent to the strainer performance analysis will be subject to future reviews of the licensee's responses to RAIs in support of GL 2004-02.

The NRC staff reviewed the licensee's assumptions and analysis with respect to post-accident containment sump pH and concluded that conservative values were used for the key parameters of the calculation. The assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment sump pH. The NRC staff finds the proposed changes acceptable with respect to their impact on post-LOCA sump pH.

3.4 Human Factors

3.4.1 Description of Operator Action(s) and Assessed Safety Significance

The required operator action is to initiate the CS, in lieu of automatic actuation. The proposed manual start of the CS is very similar to the original manual start except that the sequence is changed slightly, only one train is started, and the suction will be pre-aligned to the sump because the RWST will no longer be used as a source for the CS. The cue is the RWST low level alarm that initiates the alignment of RHR and the CS to the sump per procedure steps. This action is considered time-critical by Duke Energy and will be monitored to ensure that time available always exceeds time required.

In accordance with the generic risk categories established in Appendix A to NUREG-1764, this task sequence is considered "risk-important" due to the fact that it is required to successfully transition the plant to the sump recirculation phase of a LOCA. Because of its risk importance, the NRC staff performed a "Level One" review, i.e., the most stringent of the graded reviews possible under the guidance of NUREG-1764.

3.4.2 Operating Experience Review

The LAR is based on the initiative by the Nuclear Energy Institute and the PWR Owners Group to extend the post-LOCA injection phase and to delay the onset of the containment sump recirculation phase. Duke Energy's participation in that effort brought to bear the operating experience of the nuclear plants involved, including its sister plant, the Catawba Nuclear Station, Units 1 & 2 (Catawba 1 and 2). Industry experience was used to develop operating strategies as well as reasonable frequencies for testing and surveillances of the affected systems. The NRC staff finds Duke Energy's application of operating experience acceptable.

3.4.3 Functional Requirements Analysis and Function Allocation

Because this operator action is not a completely new action, a full functional requirements analysis and function allocation was not necessary. Prior experience with manual "restarts" had shown that operators, when allocated this task, had sufficient time and resources available to perform it reliably. If the licensee's engineering analysis had shown that the required tasks could not be done within the time constraints established, the NRC staff would have expected allocation of this function to the operators to be considered infeasible. However, this was not the case and there was no need for either a new functional requirements analysis or further consideration of allocation of function. The NRC staff finds the Duke Energy approach acceptable based on their demonstration of adequate margin to proposed time constraints, their characterization of the action as a time-critical action, and their intent to monitor the feasibility and reliability of the action over plant life.

3.4.4 Task Analysis

Because this operator action is not a new action, the only aspect requiring reanalysis was the establishment of time constraints for the action sequence. The licensee established the design value of 20 minutes for the time to initiate the CS based on peak containment pressure of 15 psig and maximum sump liquid temperature of 190 °F as stated in the UFSAR. The design values for the timing of the action sequence were later validated (see Subsection 3.4.9 of the safety evaluation, Human Factors Verification and Validation, below). The simulator testing demonstrated substantial margin to this design time. The NRC staff finds the Duke Energy update to the task analysis acceptable based on their engineering evaluation of adequate margin to proposed time constraints.

3.4.5 Staffing

Staffing and qualification are not affected by the proposed LAR. No new or additional staff are required, nor are there any new or additional qualifications required to perform the action sequence within the time constraints established.

3.4.6 Human-System Interface Design

Human-System Interface (HSI) design, including the design of the Safety Parameter Display System (SPDS) will be affected by the proposed LAR. Duke Energy has identified that the following changes are among those necessary:

- Deletion or disabling the CSS automatic actuation circuitry
- Adjust the RWST low level actuation and alarm setpoint and low-low level alarm setpoint.
- Eliminate CSS actuation from the manual Phase "B"/Containment Spray pushbutton
- Installation of a new redundant safety related wide range RWST pre-low level annunciator alarms on the existing level instrument channel
- Replacement of the existing narrow range RWST level transmitters to improve accuracy and support a higher TS minimum limit
- Elimination of CPCS automatic interlocks for CS Pump restart and re-opening of discharge isolation valves
- The Safety Parameter Display System (SPDS)

  In the original design, CS would normally be in service for any event characterized by containment pressure greater than 3 psig. Following implementation of the proposed modification, CS will not automatically start, and it is not desired to manually start the system until the suction transfer to the containment sump has occurred. An additional decision block is added such that the "ORANGE" path to FR-Z.1 is not enabled until the system has been aligned to the containment sump. A new decision box based on the status of the containment sump alignment will be added. The "ORANGE" path is only allowed if at least one train is aligned to the sump. The criteria in the decision box for determining if CS is "running" will include the pumps running and cooling flow to the containment spray heat exchanger.

These items are consistent with the Westinghouse Owners Group/Emergency Response Guidelines guidance that the status tree indicates an orange priority if CS is required, but is not operating.

- RWST Level Indication

  1. The safety related setpoint for "FWST 2/3 Lo Level" (the abbreviation "FWST" is the Duke Energy-specific nomenclature for the standard industry abbreviation "RWST") within each of the associated 7300 System protection cabinets will be adjusted to reflect the value in Table 3.1.9-1.

  2. The safety related setpoint associated with the "FWST Lo Lo Level" annunciator will be changed to reflect the Table 3.1.9-1 value.

  3. A new safety related "FWST Pre-Lo Level" annunciator will be added with a setpoint reflecting the value specified in Table 3.1.9-1.

  A human factors review of proposed changes to annunciators and status lights on the Control Room Main Control Boards and the Engineered Safety Features Bypass (1.47 Bypass) System is performed.

The NRC staff finds the Duke Energy HSI design modifications acceptable based on their identification of affected interfaces, and the use of human factors reviews and validation and verification of the design changes to confirm their effectiveness.

3.4.7 Procedure Design

The changes being made to the emergency operating procedures (EOPs) include:

a) The EOPs currently allow a portion of the RHR system flow to be aligned to an upper CS header as an independent method of providing additional containment spray flow. Auxiliary spray may be manually placed in service anytime after RHR is aligned to the sump and a minimum of 50 minutes have elapsed after reactor trip following a LOCA. Following approval of this LAR, the EOPs will be revised such that RHR auxiliary spray is manually aligned only upon reaching the containment vessel design pressure setpoint. The effect is that RHR auxiliary spray is not required because the containment pressure remains below the nominal setpoint selected in the analysis for aligning RHR auxiliary spray, i.e., the containment design pressure. The reason for maintaining the action in the EOPs is that the proposed plant EOP setpoint for manually aligning RHR auxiliary spray may be decreased in the future to accommodate plant changes or provide additional peak containment pressure margin.

b) Operator action to transfer high head and intermediate head safety injection pumps to take suction from RHR pump discharge is taken upon receipt of the RWST low-low level alarm to assure that the action is complete prior to depletion of the RWST.

c) Following transfer of the RHR pump suction to the containment sump at the RWST low level alarm, one CS Pump may be manually started after verification of adequate sump level and confirmation that containment pressure remains greater than 3 psig. This is a change from the current EOP strategy that starts both CS Pumps if they are available.

d) For certain small break size events, the non-safety-related containment ventilation units will be secured and/or isolated to avoid sump dilution and gain sump level margin by melting ice.

e) For sequences leading to containment sump recirculation, a verification of adequate sump level will be added just prior to the occurrence of switchover sump level. Direction will be added that the RHR pumps must be secured if adequate sump level does not exist.

f) Existing steps to secure containment spray when aligned to the RWST will be removed; the RWST will no longer be utilized as a suction source for CS.

g) Various steps checking general plant alignment will be adjusted to reflect changes to containment spray status.

h) RWST setpoints will be revised to reflect those values stated in Table 3.1.9-1 of the LAR.

i) The transfer to cold leg recirculation sequence will be changed as described below:

- o Verify successful autoswap of RHR suction (no change).
- o Start one train of the CS (instead of two).
- o Allow high head and intermediate head pumps to continue injection from RWST inventory until low-low RWST level.
- o Align high head and intermediate head pumps to RHR pump discharge.

j) Nonsafety-related cooling to containment will be secured for small-break LOCA scenarios when Air Return Fans are started. This gains sump level margin for small-break LOCAs.

Regarding these procedure changes, Duke Energy has committed to the following in the LAR:

> Prior to actually utilizing the provisions afforded by the approved amendments, McGuire [1 and 2 staff] will have in place all required design, document, process changes and personnel training necessary to support these provisions on the affected Unit.

Because the EOPs are an integral part of the licensed operator qualification and requalification training programs, training on the proposed action sequences will be included in both initial and continuing operator training. The NRC staff finds the proposed procedure changes acceptable based on Duke Energy's identification of affected procedures, and the validation and verification of the procedure changes to confirm their effectiveness.

3.4.8 Training Program and Simulator Design

The licensee determined that the McGuire 1 and 2 simulator is capable of modeling the proposed task sequences and will, therefore, be used in training. Training on the time-critical aspect of the task sequences will be completed prior to amending the TS. The plant modifications to the RWST, CS, and ESFAS and their impact on plant EOP response requires that a training needs analysis be completed. Duke Energy plans to complete the needs analysis, to revise training materials, and to complete training prior to implementation of the proposed LAR in training program areas such as:

- refueling Water System lesson material
- ESFAS lesson material
- CS System lesson material
- EPs lesson materials (as they relate to injection and recirculation core cooling)
- functional restoration procedures (as they relate to containment conditions during high-energy line breaks inside containment)
- simulator guides containing the above subject matter

Classroom training will typically include:

- an explanation as to the reasons this modification is being installed
- a summary of the engineering modification packages being installed
- summary descriptions on the type of accident scenarios where the RWST, CS, and ESFAS changes will impact operator responses
- a general walkthrough of the affected procedures, with an explanation of any new or modified operator tasks

Simulator training will typically include:

- information from classroom training
- accident scenarios to exercise the procedure changes, and any new or modified tasks considered to be skill-based

Duke Energy stated that the above training will be developed and presented once for all affected licensed operators, and non-licensed operators as required. Following completion, the information will be incorporated into the existing training materials and simulator guides. Based on the fact that the revised action sequences will be included in the training program and that the training changes will be implemented prior to amending the TS, the NRC staff finds that the training to be provided is acceptable.

3.4.9 Human Factors Verification and Validation

Time testing at the McGuire 1 and 2 simulator was performed to demonstrate sufficient margin to the licensee-established design values. The validation and verification of the design changes proposed by this LAR were conducted in accordance with directives provided by Duke Energy procedures. These directives describe the process used to create, check and approve engineering changes. The procedure drafts and any future procedure changes related to this LAR will be validated with actual operators to enhance operator timing and the probability of success, as is the normal practice for Duke Energy EOPs. Because the most significant change proposed by this LAR is the manual start of a CS pump in lieu of the automatic start of both CS pumps, a sensitivity study was performed to evaluate the containment pressure response to a CS Pump failure. The results of this sensitivity study indicate that 20 minutes are available for operator action to initiate CS spray flow from the other CS pump. If CS flow is initiated within 20 minutes, the peak containment pressure will remain below the containment design pressure. Duke Energy found that the operator action is achievable within the 20 minute timeframe.

The NRC staff reviewed the human factors verification and validation associated with this LAR and found it acceptable based on the fact that operator action has been verified and validated to be achievable within a 20 minute timeframe and during that timeframe the peak containment pressure will remain below the containment design pressure.

3.4.10 Human Performance Monitoring Strategy

The actions proposed by this LAR will be included in the licensee's program for monitoring time-critical tasks and sequences. This will provide a means to: a) ensure that the time-critical actions within the scope of a procedure can be accomplished by plant personnel, b) document periodic validation of credited action times, and c) ensure that subsequent changes to the plant, procedures, or programs will not invalidate the credited action times. Based on the administrative protection against inadvertent change and the periodic re-validation provided by the licensee's monitoring program, the NRC staff finds the Duke Energy human performance long-term monitoring strategy acceptable.

3.4.11 Summary of Human Performance Evaluation

Based on the evidence provided by Duke Energy, i.e., that sensitivity-testing in the simulator demonstrated significant margin to design, as well as the appropriate administrative controls being applied to procedures, training, and HSI design, and the application of industry and in-house operating experience, the NRC staff concludes that the proposed LAR is acceptable with respect to the topic of human performance.

3.5 Design Basis Accident Radiological Consequences

The NRC staff reviewed the licensee's regulatory and technical assessments, as they relate to the radiological consequences of the DBA analyses, in support of the proposed LAR. The licensee determined that the only DBA analyses which are affected by the manual initiation of CS to mitigate post-accident radiation doses are the design basis Main Steam Line Break (MSLB) and LOCA accidents. Only the postulated MSLB accident and LOCA analyses credit the CS for accident mitigation. This section of the evaluation focused on the specific effect of the proposed LAR, while ensuring consistency with the licensee's current AST assumptions which determined the design basis LOCA to be limiting for the accident analysis. Information regarding the effect of the LAR on the design basis LOCA dose consequence analysis was provided by the licensee in Attachment 1 to the LAR. The findings of this safety evaluation are based upon the descriptions and results of the licensee's assessment and other supporting information docketed by the licensee, in particular the UFSAR and the AST amendment (Reference 4).

The reanalysis of the containment response to a large-break LOCA has been determined using the Duke Energy ice condenser containment response methodology described in Topical Report DPC-NE-3004-PA, which was approved by the NRC staff in Reference 3 as part of the license amendments that implemented the ECCS water management initiative at Catawba 1 and 2. The containment response parameters of temperature, pressure, and air flow are all inputs that are considered in the dose analysis. McGuire 1 and 2 are both ice condenser containments equipped with feedring steam generators (FSGs). FSGs are limiting for containment response due to higher initial primary containment and secondary containment fluid mass. In addition, the response of the McGuire 1 and 2 ice condenser containment following a DBA has a direct effect on the resulting radiological consequence dose values. Duke Energy has previously reanalyzed Catawba 1, a similar ice-condenser containment site which is also equipped with an FSG, and the revised analysis was used to establish a new baseline before performing the sensitivity study to support similarly LAR approved by the NRC staff in Reference 3. The containment response methodology in Reference 3 applies to McGuire 1 and 2 and there is a close similarity between the analyses for Catawba 1 and McGuire 1 and 2. Therefore, the NRC staff finds the licensee's proposed use of the contained response methodology, along with the analysis of Catawba 1 and the associated FSG, approved by the NRC staff in Reference 3 to be acceptable for application to McGuire 1 and 2.

The radiological consequence analysis of the design basis LOCA assumes full core melt with release of the radioactive material to the RCS and then to the containment over a total period of 2 hours. The source term used by the licensee assumes full power operation, including calorimetric uncertainty, to give 102 percent of rated power. The NRC staff has reviewed and accepted the licensee's determination that the proposed LAR affects six parameters (inputs) in the radiological consequence analyses of DBAs at McGuire 1 and 2 that have been previously reviewed and approved by the NRC staff in Reference 4:

- Spray Lambdas
- Sump pH
- the timing of the exhaust and recirculation cycles of the Annulus Ventilation System
- the time dependent sump level (volume) profile
- the time that ECCS back-leakage begins
- partitioning factors for ECCS releases to the RWST and to the Auxiliary Building

The licensee indicates that all other features of the model and/or parameters used in the analysis are unaffected, including the source term, the source term release timing, the response to the Containment Air Return system (CAR), containment leakage modeling, control room ventilation system modeling, atmospheric dispersion, ECCS leakage rates, filtration models, and release locations.

The NRC staff has reviewed and accepted the licensee's determination that all other features of the model and/or parameters used in the analysis are unaffected. The following sections address the impact of the licensee's proposed LAR on the above-stated six affected parameters.

3.5.1 Containment Airborne Radioactivity (Spray Lambdas)

Airborne radioactivity removal by the CS is modeled using removal constants referred to as "spray lambdas." The spray lambdas model the ability of the CS to remove elemental and particulate iodines from the upper containment atmosphere in a post-LOCA environment. The licensee derived spray lambda values using inputs related to the characteristics of the containment, the flow characteristics of the CS, and, during recirculation, the chemistry (e.g. pH) and temperature profiles of the containment sump fluid. Currently, as indicated in Section 3.1.1.2.2 of Reference 4, Duke Energy does not assume credit for fission product removal by sprays until the start of the CAR fans, which is consistent with the assumption that the fission products are initially released in the lower compartment which has no spray. As shown in Table 3.2.8-4 of the LAR, the licensee takes credit for elemental iodine removal by the CS until a decontamination factor (DF) of 200 is reached, which was previously calculated to occur at 46,000 seconds after accident initiation. With the proposed LAR, the DF of 200 will be reached at 104,000 seconds after accident initiation, extending the overall effectiveness of the CS for elemental iodine removal. Once the DF exceeds its limit, consistent with RG 1.183, the effectiveness of the CS in removing the iodine from the containment atmosphere is reduced. Therefore, in accordance with RG 1.183, the licensee also assumes that the particulate spray removal effectiveness of the CS removal is reduced to 10 percent of the calculated value (DF of 20) for the duration of the spray, considering that a DF of 50 would be reached very soon after the initiation of the CAR fans. The licensee previously determined that this reduction will occur at 7,100 seconds after accident initiation. However, Table 3.2.8-1 of the LAR indicates that the particulate iodine DF of 50 will be reached at 8,000 seconds after accident initiation after the proposed LAR is implemented. The licensee did not credit the removal of organic iodines from CS operation. The NRC staff reviewed the licensee's modeling of fission product removal in containment and determined that it is acceptable as it follows the guidance in RG 1.183.

The McGuire 1 and 2 CS has two trains of safety-related pumps, heat exchangers, upper containment spray nozzles, and associated valves and piping. Duke Energy's analysis of record (AOR) does not take credit for the scrubbing of radioactive materials in the ice condenser; however, as a design basis, Duke Energy credits the chemical and physical properties of the ice condenser ice as an iodine removal system to reduce the fission product iodine concentration in the post-LOCA containment atmosphere. Following approval of this LAR, the CS automatic start signal will be removed and spray operation will be manually controlled when pump suction is aligned to the containment sump. Thus, the licensee's evaluation takes credit for the CS to reduce containment pressure, temperature, and radioactivity in the upper containment compartment, and for removal of fission products other than organic iodine and noble gases by the CS during the cold leg recirculation and hot leg recirculation phases of the accident.

The time constants and DF cutoff times for washout of fission products with the McGuire 1 and 2 CS were revised to account for the proposed LAR. With the proposed LAR implemented, it is assumed that the operators complete the alignment of the CS to the containment sump and start only one CS pump for post-LOCA recirculation at 4,800 seconds (80 min.), no simulation of RHR system alignment to the auxiliary CS headers, and start of only one CS pump regardless of the design basis LOCA scenario. The time constants and cutoff times, representative of the proposed LAR for the baseline analysis, are compared in tables found at the end of this safety evaluation in Table 1 for diatomic iodine and Table 2 for fission product particulates. The methodologies used by Duke Energy to calculate the time constants for elemental iodine removal and particulate removal by sprays follow the general guidance found in RG 1.183, as well as the more specific guidance found in Section 6.5.2, "Containment Spray as a Fission Product Cleanup System" Revision 2, of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, [light water reactor] LWR Edition." The tables indicate that in the baseline analysis, it is conservatively assumed that the operators stop the CS pumps at 3,000 seconds (50 min.) for the design basis LOCA with minimum safeguards (one CS pump in operation) and at 1,530 seconds (25.5 min.) for all other design basis LOCA scenarios. It is also assumed that for all design basis LOCA scenarios at 3,000 seconds, the operators start auxiliary

CS from one RHR pump. In addition, the entire RCS inventory was assumed to be at TS levels and released to the containment at T = 0 hours. It is also assumed, in accordance with the guidance of RG 1.183, that 100 percent of the airborne activity is instantaneously and homogeneously mixed in the containment atmosphere.

Consequently, when the time constants for washout of particulates are applied to spray washout, the DF for spray removal of particulates was determined by the licensee to reach 50 for the LOCA, both with one and/or two CS pumps in operation. As shown in Table 3.2.8-4 of the LAR, the particulate spray lambdas are reduced by a factor of 10 for reduced spray effectiveness due to particulate spray washout at DF of 50. In the previously approved baseline analysis (for Catawba 1 and 2), a DF of 50 was obtained at 7,100 seconds. The LAR also indicates that spray washout for elemental iodines reached a DF of 200 at 46,000 seconds in the baseline case, whereas this DF was reached at 104,000 seconds in the proposed LAR. The note on Table 3.2.8-4 of the LAR indicates that this is predicted to occur after spray ceases to be credited at 24 hours, so full elemental removal credit is not taken. Spray is not credited for any iodine removal after 24 hours (86,400 seconds). In the CLB, spray is modeled to start at 120 seconds, but not credited until 600 seconds when CAR fan starts. Thus, with the proposed changes in place, the DF for spray removal of particulates was calculated by the licensee to reach 50 for all design basis LOCA scenarios.

The NRC staff reviewed the analysis of the licensee's proposal for the operator to manually initiate only one CS pump to mitigate a LOCA and has concluded that it is acceptable based acceptable results with regard to removal of containment airborne radioactivity and on the consistency of its methodology with RG 1.183 and section 6.5.2 of NUREG-0800.

3.5.2 Annulus Ventilation System (AVS)

Section 6.2 of the UFSAR describes the containment as consisting of a containment vessel and a separate reactor building enclosing an annulus air space. Both McGuire 1 and 2 units also have ice condenser containments. The ice condenser is used to limit containment pressure and temperature following a LOCA. This containment is divided into a lower compartment, an upper compartment, and the ice condenser. The reactor and RCS are located in the lower compartment. Following a LOCA, steam and/or airborne fission products are directed through the ice condenser and condensed to limit containment pressure. Condensed steam and/or airborne fission products and melted borated ice are routed into the lower containment by the ice condenser floor drains. As described in Section 6.1.5 of the UFSAR, the melted borated ice provides a suitable inventory of borated water for the containment sump, which is subsequently used to control the containment sump pH levels. The only connections between the lower and upper compartments are the CAR fans, which blow air from the upper to the lower containment, and the ice condenser, through which the LOCA blowdown passes from the lower to the upper containment.

The CAR works to promote the exchange of the atmospheres of the upper and lower containment compartments by returning air from the upper containment to the lower containment. The air in the lower containment is forced through the ice condenser and into the upper containment where the spray system can remove the activity deposited in the upper containment atmosphere. Conservatively, the licensee did not credit activity removal by the ice condensers during this process. Specifically, potential natural air flow through the ice condensers and into the upper containment prior to the CAR start as a result of the thermal conditions in lower containment was not credited. The licensee modeled the AVS to start and run as previously analyzed. However, the change in the thermodynamics associated with the LAR LOCA response modifies the durations of the exhaust and recirculation cycles. The licensee conservatively modeled the CAR to start 10 minutes after accident initiation, which is the latest possible time based upon worst case diesel generator loading. Therefore, the licensee did not credit CS activity removal until CAR system initiation. However, since the gap release from the fuel is not assumed to occur until 30 seconds after the initiation of the event, the blowdown phase is not assumed to include the LOCA fission product source term. Also, Duke Energy assumed that the fission product source term is released to, and mixes homogeneously in, the lower compartment volume only. The licensee's modeling of the LOCA release into containment and subsequent containment mixing is in accordance with RG 1.183.

The secondary containment, or annulus, is provided with the AVS which provides a vacuum in the annulus to promote air flow from adjacent higher pressure spaces into the annulus where it can be held and filtered prior to being released to the environment. In response to a LOCA, the AVS initiates in the exhaust mode, discharging through the filters, to reduce pressure in the annulus. The AVS is activated by a safety injection (SI) signal, and draws the annulus to a negative pressure such that any primary containment leakage will flow into the annulus volume. The system reduces the concentration of airborne activity within the annulus and it filters any air discharged from the annulus to the environment.

Consistent with how the design basis LOCA was previously analyzed in Section 3.2.8 of Reference 4, Duke Energy reanalyzed the post-LOCA conditions in the annulus and the AVS response with the containment pressure and compartment temperatures associated with the proposed LAR for the three following DBA LOCA scenarios:

1) DBA LOCA with minimum safeguards - only one of the two trains responds as designed.

2) No AVS failures, with failure of a CS or residual heat removal system heat exchanger.

3) Failure of one AVS pressure transmitter - one train operates in exhaust mode until operators secure it 2.5 hours after the initiating event.

However, for the reanalysis of the DBA LOCA, Duke Energy changed its analysis of the partitioning of containment leakage. In the reanalysis the containment leakage was partitioned by source, 60 percent from the lower compartment and 40 percent from the upper compartment. Prior to this reanalysis, the containment leakage was partitioned in proportion to the volumes of the lower and upper compartments. As a complementary measure, containment leakage to the annulus was assumed to mix with half of the air in the annulus, then filtered for release by the AVS.

The post-LOCA annulus drawdown times and AVS exhaust and recirculation airflow rates, for the DBA LOCA scenarios limiting for offsite and control room radiation doses and with the proposed LAR in place, were all analyzed by Duke Energy in Section 3.2.8.7 of the LAR. The time for the AVS to draw the annulus pressure to -0.25 inch water gauge (inwg) everywhere inside the annulus and the AVS exhaust and recirculation airflow rates were used as inputs to the AST analysis of the design bases LOCA in support of the proposed changes. In Section 3.2.8.7 of the LAR, the licensee has shown that, after implementing a new containment analysis, based on the

CS and/or RHR auxiliary CS not being actuated for the LOCA, the containment pressure still decreases enough by 24 hours after the accident initiation to assume the 50 percent reduction in leak rate.

For the proposed changes, the licensee modeled the pressure required for the establishment of a vacuum condition in the annulus. Thus, the modeled annulus pressure value for establishment of annulus vacuum was changed from -0.90 inwg for the CLB, to -0.97 inwg for the reanalyzed condition consistent with the proposed LAR. In comparing the modeled annulus pressures for both the CLB and the proposed LAR, it was determined that the impact of the proposed changes will be minimal based on the resulting pressures for the AVS model (a -0.07 inwg difference).

The NRC staff evaluated the licensee's analysis with respect to annulus mixing and AVS filtering. Taking the minimal -0.07 inwg difference in model pressures into consideration, along with the predicted depletion times associated with the LAR, as presented in Table 3.2.8-5 of the LAR, the NRC staff has reasonable assurance to believe that the assumptions on the annulus mixing and percentage filtered by the AVS are acceptable and consistent with the McGuire 1 and 2 CLB.

3.5.3 ECCS Back-Leakage

Consistent with the approved AST analysis in Reference 4, the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that, except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of five percent released to the containment sump water during the gap release phase and 35 percent released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ECCS systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections.

In addition to the leakage of radioactive iodine in molecular form directly from the containment atmosphere to the outside, Duke Energy identified a possible leakage path through the ESF in its AST analysis. The containment sump water can leak either to the RWST or to the Auxiliary Building. Water can leak to the RWST through the valves that isolate the ESF recirculating system from the RWST. Since, after an accident, this water will contain radioactive iodine and since the RWST is vented to the atmosphere, this back leakage will constitute a path for the radioactive iodine to leak to the outside and contribute to the external dose rates. In the LAR, Duke Energy reanalyzed this source of iodine release for a LOCA with the proposed LAR changes assumed.

In contrast to the analysis approved by Reference 3, a slight modification to the single failure impact to the ECCS back-leakage model was made in the LAR for conservatism; however, the analysis resulted in an inconsistency between the ECCS release model and the containment release model. In the Minimum Safeguards scenario, it would logically follow that one train of Chemical and Volume Control System, RHR, and SI would respond to the LOCA. For the purpose of determining the time that ECCS back-leakage starts (time of RWST low level), and for this purpose only, the licensee assumes that two trains of these systems respond within the docketed analysis for the proposed LAR. Modeling two trains conservatively empties the RWST more rapidly, which causes the ECCS back-leakage releases to start earlier, which in turn adds additional conservatism to the licensee's current radiological consequence AOR.

The NRC staff has reviewed the licensee's evaluation of ECCS back-leakage and finds it acceptable in that the deterministic approach of RG 1.183 was used and conservative assumptions were made regarding depletion of the RWST and onset of ECCS back-leakage.

3.5.4 Sump pH

In developing the LAR analysis of sump pH, Duke Energy maintained consistency between the Catawba 1 and 2 and McGuire 1 and 2 models. The licensee indicates in Section 3.2.8.1 of the LAR that this effort resulted in a small increase in the amount of conservatism relative to what a specific analysis for McGuire 1 and 2 would have yielded. The partitioning factors associated with the ECCS releases are taken from the LOCA analysis approved in Reference 3, as they were for the baseline (i.e. CLB) case. The iodine and particulate partitioning factors are closely correlated to the sump via the measured pH level of the sump water. In regard to the proposed LAR, the licensee indicates that the physical properties of the plants have not changed; however, the post-LOCA pH and sump temperature responses have changed. Although in the post-LOCA conditions the sump water pH is maintained at the value of equal to or higher than 7, when it mixes with the water remaining in the RWST after the injection phase, its pH could drop to a value significantly below 7. Low pH favors conversion of the dissolved iodine from a soluble ionic form to the scarcely soluble elemental form. Some of this iodine will be, therefore, released to the RWST air space from which it could leak to the atmosphere. The licensee assumed that the sump water leaks into the RWST at a rate of 10 gpm. As more of this water leaks into the RWST, both its pH and the concentration of iodine increase.

In Section 3.2.8.3 of the LAR, the licensee indicates that the minimum pH is essentially the same for both plants (the McGuire 1 and 2 pH is slightly higher than that of Catawba 1 and 2 by about five hundredths of a pH unit) and the equilibrium-corrected McGuire 1 and 2 post-LOCA pH is slightly higher (about six hundredths of a pH unit) than that of Catawba 1 and 2. The post-LOCA sump pH analysis sump equilibrium temperature used for the Catawba 1 and 2 analysis (185° F) is higher than that predicted for McGuire 1 and 2 (177° F). The McGuire 1 and 2 sump pH continues to be slightly higher than that of Catawba 1 and 2, while the sump equilibrium temperature continues to be slightly lower, so the Catawba 1 and 2 post-LOCA partitioning model input continues to be limiting and it can continue to be conservatively adopted for the McGuire 1 and 2 LOCA analysis previously approved by the NRC staff in Reference 4.

The NRC staff has reviewed the licensee's analysis of sump pH and found it acceptable as it uses the post-LOCA partitioning model from Catawba 1 and 2 approved in Reference 4 to obtain more conservative results for use in the LOCA analysis for the LAR.

3.5.5 Sump Level Profile

The proposed LAR employs a time-dependent sump profile derived for the LAR scenario. The licensee indicates that the sump volume builds early in the accident, peaks, and declines slowly, as presented in Table 3.2.8-6 of the LAR. The model value chosen for each time period is equal to or lower than the predicted values for that entire period. The licensee further indicates in

Section 3.2.8.8 of the LAR that the chosen model conservatively bounds the sump level prediction over the time period because a lower volume will produce a greater activity concentration. Furthermore, a steady state value is conservatively chosen to be lower than the volume at the end of the curve. Based on the level of conservatism that exists in the sump level profile model, the NRC staff determined that the bounding time-dependent sump volume model, which is based upon the expected plant response and used in the reevaluated radiological consequence analysis, was conservatively modeled by the licensee.

3.5.6 Partitioning Factors for ECCS Back-Leakage to the RWST and Auxiliary Building

Increased iodine partition fractions for ESF backleakage to the RWST were calculated with the initial water volume set to correspond to the RWST ECCS outlet elevation and vortex allowance. Consistent with what was approved by the NRC staff in Reference 3, and in contrast with the 20 gpm backleakage rate which is currently assumed in the AST analysis for McGuire 1 and 2 (Reference 4), the leakage into the RWST for this analysis is assumed to be 10 gpm. The source term for the ECCS leakage is the radioactivity in the containment sump water which is assumed to consist of only the iodine isotopes and their precursors. The iodine species in the ECCS leakage release are assumed to be 97 percent elemental iodine and 3 percent organic iodine compounds consistent with guidance in RG 1.183. Duke Energy's methodology for calculating the McGuire 1 and 2 site-specific iodine release fractions (partitioning) from the ECCS leakage was previously approved in Reference 4.

Pursuant to the Catawba 1 and 2 post-LOCA partitioning model, Duke Energy based the calculation of the McGuire 1 and 2 iodine partitioning on the methodology in NUREG/CR-5950, "Iodine Evolution and pH Control." Using this methodology, the licensee calculated the formation of volatile elemental iodine based on the concentrations of iodine and iodide ions in the leakage. The total amount of iodine in the containment sump water, including stable I-127 and long-lived I-129, was used in the pH calculations. Post-LOCA containment sump pH was calculated with the proposed LAR changes assumed to be in place. The water inventory of the RWST was set to its upper bound value, adjusted only for the elevation of the outlet piping. In particular, no flow from the CS was simulated. Flow from the ECCS was used to simulate the transport of boric acid solution from the RWST to the containment sump. The CS was not modeled to take suction from the RWST in accordance with the proposed changes. In addition, the iodine partition fractions for ESF backleakage to the RWST were recalculated based on changes to the containment sump pH and RWST low-low level setpoint. Duke Energy also included some conservatism in regard to its ECCS leakage analysis. For each ESF leakage scenario and time interval, the recalculated value was compared to the baseline (i.e, Catawba 1 and 2) values. The higher of the two values was taken for the iodine partition fraction. All other input to the calculation of post-LOCA containment response remained unchanged from the CLB.

The NRC staff evaluated the licensee's ECCS Leakage and pH calculation as discussed above in Sections 3.5.3 and 3.5.4, respectively, of the safety evaluation as it pertains to post-LOCA particulate release to the containment environment. The licensee indicates in Section 3.2.8.3 of the LAR that the same methodology used in the Catawba 1 and 2 and McGuire 1 and 2 CLB analyses was used for the proposed LAR modeled partitioning of ECCS leakage to the Auxiliary Building. The licensee further indicates that this partitioning model was approved by the NRC staff in Reference 3. The licensee calculated time-dependent water temperatures, iodine concentrations in the liquid and vapor, and the water pH for both the ECCS fluid and RWST, of which the water pH for the ECCS fluid is limiting. As was done in the McGuire 1 and 2 CLB analysis, the partitioning factors were adopted without examination as to the case most applicable to McGuire 1 and 2. Instead, as additional conservatism, the set of partitioning factors which results in the largest releases were adopted by Duke Energy. The relative relationship between the two sites for these sump parameters remains the same. The modeled release rate is 1 gpm, which is twice the operational leakage rate. The difference between the release fractions is that the proposed change results in the second time-step being slightly greater than the current analysis, but both sets are nearly identical.

Using these calculated values, Duke Energy determined iodine release rates from the RWST atmosphere to the environment and ECCS leakage areas to the environment. The limiting iodine partition fractions for the post-LOCA ESF leakage are shown in Tables 3 and 4 of this safety evaluation. The values associated with the proposed LAR are compared to the baseline values in these tables, and the time to begin recirculation with the proposed LAR changes in place was set to 2160 sec (0.6 hrs). Specifically, Table 3 indicates that the iodine partition fractions for post-LOCA ESF leakage presented correspond to the LOCA scenarios limiting for radiation doses at offsite locations and in the control room. The licensee indicated that the filters of the Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) are aligned to the ESF pump rooms; which is assumed in the analysis to occur after 72 hours. The proposed changes increases the iodine partition fraction to 0.004, as presented on the bottom of Table 3 (modeled LOCA with failure of cooling water flow through a RHR or CS Heat Exchanger).

Table 4 of this safety evaluation indicates that the partitioning factors for the RWST releases are very low and that the impact from RWST releases, ultimately, is not substantial. As indicated, from the time of sump recirculation until two hours, the RWST iodine partitioning factors will be lower with the proposed LAR in place because of the later start of recirculation and ECCS back-leakage. By two hours, the impact of the smaller initial tank inventory and higher sump water temperature causes the newly proposed RWST iodine partition fractions to become larger than the baseline values. As a result, the back-leakage rate to the RWST remains 20 gpm, which begins at 2160 seconds, as indicated in Table 4 and as discussed above.

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the partition fractions of the iodine dissolved in sump water converted into the elemental form. Based on the results of these calculations, the NRC staff determined that the proposed changes resulted in a negligible fraction of the dissolved iodine being converted into elemental form and a low release of radioactive iodine to the environment. Therefore, the NRC staff concludes that the amount of iodine that would be released from the RWST and the auxiliary building as a result of backleakage would be negligible because of the low concentration of iodine when the tank pH is below 7.

3.5.7 Control Room Habitability and Modeling

The design basis LOCA dose consequence analysis modeling for the McGuire 1 and 2 control room design and operation would change as a result of the proposed LAR modifications. The calculation of post-LOCA radiation dose accounts for release of activity, transport of effluent to the Control Room Area Ventilation System (CRAVS) outside air intakes, and buildup of activity in the control room. The CRAVS ensures that the control room will remain habitable for personnel during and following all credible accident conditions. As described in Section 3.1.11 of the LAR, this function is accomplished by the Outside Air Pressurization Filter trains which pressurize the control room to greater than 1/8 inwg with respect to the exterior with filtered outside air. The system consists of two independent, redundant trains of equipment with each train consisting of a pressurizing filter train fan, filter unit, and associated dampers and duct work. Upon receipt of an ESF signal, both trains are currently credited to start.

Consistent with the description in Section 3.1.1.6 of Reference 4, Duke Energy assumed that the CRAVS initiates at the time of the core damage, and begins pressurization and filtration of the control room. Duke Energy also assumes that one train of the CRAVS is immediately secured by the operators; therefore, only one train of CRAVS is credited in the dose analysis. The limiting value for asymmetry in the airflow into the two CRAVS outside air intakes was determined to be 65/35, and the control room atmospheric dispersion factors (χ/Q) were adjusted to account for the imbalance. Also consistent with the AST analysis approved in Reference 4, the licensee assumed the lower limits for the CRAVS filter train recirculation flow rates. The licensee also calculated doses for both the lower limit and upper limit for the CRAVS intake flow rate. The licensee assumed that the unfiltered inleakage into the control room was 625 cubic feet per minute (cfm) prior to pressurization, and 210 cfm after pressurization for the duration of the event.

Other contributors to the post-LOCA control room radiation doses were assessed. In particular, the direct radiation dose to the control room from fission products outside the control room was calculated consistent with the regulatory positions for McGuire 1 and 2's full scope implementation of AST methodology. According to the licensee's analysis, this direct constituent to the radiation dose in the control room was unchanged from McGuire 1 and 2's current AOR and remains 1.26 rem. This value was added to the total effective dose equivalent (TEDE) from post-LOCA transport activity to the CRAVS outside air intakes to yield the total post-LOCA TEDE in the control room. The above noted changes yielded a moderate increase in the post-LOCA TEDE at the exclusion area boundary (EAB). The cumulative impact of these changes on the low population zone (LPZ) dose is small (roughly 0.3 rem TEDE).

The baseline dose at the EAB, LPZ and in the control room for the limiting LOCA scenarios are presented in Table 5 of this safety evaluation. Duke Energy then took these baseline values, applied the proposed LAR changes (modeled as the design basis LOCA, with failure of cooling water flow through a heat exchanger of either the RHR system or the CS), and compared the dose results of the proposed LAR changes to that of the baseline analysis. The latter comparison is also presented in the right-hand section of Table 5 of this safety evaluation. Duke Energy's analysis showed that either the RHR system or the CS was verified to be limiting for dose at the offsite locations, and the design basis LOCA, with an initially closed CRAVS outside air intake, was verified to be limiting for the control room dose. Therefore, the NRC staff concludes that for an assumed 2-hour duration, the total dose of the proposed LAR changes increased minimally for the EAB, LPZ, and Control Room. Specifically, the total EAB dose increased by 2.77 to 10.92 rem TEDE; the total LPZ dose increased by 0.51 to 2.70 rem TEDE; and the total Control Room dose increased by 0.61 to 4.86 rem TEDE, all of which are acceptable by NRC staff and remain consistent with the applicable dose acceptance criteria described in RG 1.183.

3.6 Component Analyses Evaluation

The NRC staff reviewed Section 3.2.3, "Component Analysis," of the LAR, and requested additional information regarding the summary of results for the CS piping stress analysis, CS piping support re-qualification, and CS modified supports in the reactor building annulus area. By letters dated March 23, 2011, and May 2, 2011, the licensee provided the requested summaries, as discussed below, to address the RAI questions.

The licensee conservatively chose a bounding value of 200° F for the sump temperature to analyze the CS piping in the auxiliary and reactor buildings, and demonstrated that the maximum thermal stresses in piping are below the allowable stress, in accordance with equations 10 and 11 of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section III, Subsection NC. The licensee, in its response, provided its results for the re-qualification of several pipe supports, which included the use of a higher sump temperature and consideration of water hammer and seismic loading, acting concurrently, in the revised piping stress analysis for the CS piping in the reactor building; and compared the results with the design basis calculations. The NRC staff finds that the licensee's justifications sufficiently address the number of pipe supports requiring re-qualification. In its response, the licensee also provided summaries for the qualification of pipe support components, demonstrating that the support component loads, stresses, fillet weld sizes, beam deflections, and interaction ratios for anchor bolts are below the respective allowable values. The licensee also identified and addressed pipe support restraints, namely three supports in the McGuire 2 reactor building annulus and one pipe support in the McGuire 1 reactor building annulus, that required modification.

Based on a review of the LAR, and responses to the RAI questions, the NRC staff concludes that the licensee has adequately evaluated the impact of the increase in sump temperature on the affected piping, components, and supports. In addition, the NRC staff concludes that the licensee has adequately demonstrated the structural integrity of the affected piping, components, and supports, by demonstrating that the computed stresses, loads, deflections, and weld sizes are below the applicable code or criteria limits. Therefore, the NRC staff finds the licensee's component analyses acceptable.

3.7 Electrical and Power Systems Evaluation

3.7.1 Environmental Qualification

The NRC staff reviewed the LAR for changes to Environmental Qualifications (EQ). In Section 3.2.3, of the LAR, the licensee states that the proposed delayed actuation of the CS can cause sump temperature to exceed current design temperature of 190 °F (up to 197 °F) for about 9 minutes. By letter dated November 15, 2010, in a response to an NRC staff RAI question requesting a supplementary discussion of the effects of increased sump temperature on EQ equipment, the licensee stated:

> In terms of the McGuire 1 and 2 ECCS Water Management Project, the increased sump liquid temperature was included in the environmental temperature profile used to evaluate EQ equipment located inside the Reactor Building (Containment and Annulus locations).
>
> The evaluation of EQ equipment is documented in a site calculation. The following information specifically addressing the sump temperature is taken from the calculation:
>
>> As shown in Figure B-4A and Figure B-4B, there are some periods of elevated temperature in the response profile calculated in Reference 6 [the Long-Term Containment Response - Manual Containment Spray Initiation] versus the current design basis profile. By including the sump temperature profile due to the ECCS Water Management modifications into the overall "EQ Equip Evaluation Curve" provided in Figure B-6A, there is additional thermal content added to the evaluation curve which provides a bounding temperature profile.

> (Note to Reviewer: Attached Figures B-4A and Figure B-4B are the logarithmic and linear scale plots, respectively, of the same data).
>
> Based on the review and evaluations documented within the calculation, there were no adverse impacts on equipment due to the revised profiles, and all EQ equipment located within the Containment and Annulus locations for McGuire 1 and 2 remain qualified for the environmental profiles proposed for the McGuire 1 and 2 ECCS Water Management Project.

The NRC staff has reviewed the LAR and the supplement dated November 15, 2010, and finds that the EQ equipment complies with the requirements of 10 CFR 50.49 and that the proposed modifications have no impact on the existing qualifications of these components. Hence, the NRC staff concludes that this EQ equipment remains qualified for their respective applications.

Figure B-4A in Attachment 1 of the letter dated November 15, 2010, "Long Term EQ Containment Temperature Response Sump Water Temperature (logarithmic time scale)," indicates that there are some periods when the calculated long-term sump water temperature is elevated above the current design basis temperature. This sump temperature rise is due to the proposed delayed actuation of the CS, discussed above. The overall EQ equipment evaluation curve in Figure B-6A in Attachment 1 of the letter dated November 15, 2010, "Long Term EQ Reactor Building Temperature Response EQ Evaluation Curve (logarithmic time scale)" includes this elevated sump temperature.

The NRC staff reviewed the LAR and the figures provided in the response to the RAI and finds that all EQ equipment located within the Containment and Annulus locations for McGuire 1 and 2 continue to follow the requirements of 10 CFR 50.49 and remain qualified for their proposed environmental profiles.

3.7.2 Evaluation of Impacts on Electrical Power Systems and Electrical Power Systems Testing

In a letter dated November 15, 2010, the licensee responded to an RAI question from the NRC staff regarding the impact of manually starting the CS pump on the emergency diesel generator (EDG) when it is fully loaded, versus earlier as part of the current automatic sequence. The licensee stated:

> As shown by the voltage and frequency traces developed in the ETAP [Electrical Transient Analysis Program] EDG dynamic analysis, the start of the containment spray pump motor at the end of the loading sequence will not have any impact on EDG operation. The frequency dip and overshoot remains within +/- 2% of nominal frequency. The voltage dip and overshoot remains within +/- 10% of nominal voltage. Therefore, the Regulatory Guide 1.9, Revision 3, restoration time to steady state is not challenged.

Based on review of the LAR and the response to the RAI, the NRC staff finds that the proposed changes to McGuire 1 and 2's TS will not negatively affect the EDG system and will continue to comply with GDC 17 requirements.

In its letter dated November 15, 2010, the licensee made the following regulatory commitment:

The EDG load sequencer logic and load acceptance testing will be completed prior to the first entry into Mode 4 operations following the implementing refueling outage.

The diesel generator load sequencer logic testing will be a temporary test procedure that will ensure the CS pump motor load shed logic operates correctly. The CS pump motor start logic is blocked during EDG Load Sequencer actuation and is not blocked following EDG load sequencer reset. The diesel generator loading acceptance testing will periodically test each EDG to ensure that they are capable of starting the containment spray pump motor after the other loads have been started. After reviewing the LAR and the proposed load sequencer logic and load acceptance testing described in the response to the RAI, the NRC staff finds that the proposed tests follow the GDC 18 requirements for periodic inspection and testing and are acceptable.

4.0 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.

5.0 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 (75 FR 61524). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The 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.

7.0 REFERENCES

1. Letter, M. S. Tuckman, Duke Energy, to Document Control Desk (DCD), NRC, "McGuire Nuclear Station, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station Docket Numbers 50-413 and 50-414, Issuance of the Approved Versions of Topical Report DPC-NE-3004-PA, Revision 1, and DPC-NE-3004-A, Revision 1; Mass and Energy Release and Containment Response Methodology TAC Nos. MA5511, MA5512, MA5517 and MA5518," December 18, 2000, (ADAMS Accession Nos. ML003780723 and ML003781040).

2. Letter, J. R. Morris, Duke Energy, to DCD, NRC, "Duke Energy Carolinas, LLC (Duke), Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Technical Specifications (TS) and/or Bases Sections: 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, 3.3.3, Post Accident Monitoring (PAM) Instrumentation, 3.5.4, Refueling Water Storage Tank (RWST), 3.6.6, Containment Spray System License Amendment Request for Emergency Core Cooling System (ECCS) Water Management Initiative," September 2, 2008 (ADAMS Accession No. ML082490094).

3. Letter, J. H. Thompson, NRC, to J. R. Morris, Duke Energy, "Catawba Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Technical Specification Changes to Allow Manual Operation of the CSS (TAC Nos. MD9752 and MD9753) June 28, 2010, (ADAMS Accession No. ML092530088).

4. Letter, J. F. Stang, NRC, to B. H. Hamilton, Duke Energy, "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Adoption of the Alternate Source Term Radiological Analysis Methodology (TAC Nos. MD8400 and MD8401)," March 31, 2009, (ADAMS Accession No. ML090890627).

5. Letter, J. F. Stang, NRC, to B. H. Hamilton, Duke Energy, "McGuire Nuclear Station, Units 1 and 2, Request for Additional Information (RAI) Regarding Supplemental Responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors" (TAC Nos. MC4692 and MC4693)," November 18, 2008, (ADAMS Accession No. ML083080350).

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Date: September 12, 2011

Table 1
Time Constants for Containment Spray Removal of Diatomic Iodine
(One Train of Containment Spray with Minimum Safeguards Failure)

Time Span (sec)		Time Constant (hr^{-1})	
		Baseline	Proposed
Start	End	CLB	Modifications
0	600	0.00	0.00
600	3,000	20.00	0.00
3,000	3,240	0.22	0.00
3,240	3,500	0.50	0.00
3,500	4,000	0.53	0.00
4,000	4,500	0.56	0.00
4,500	4,800	0.58	0.00
4,800	5,000	0.58	0.27
5,000	7,100	0.59	0.27
7,100	8,000	0.59	0.27
8,000	24,600	0.59	0.27
24,600	30,000	0.58	0.27
30,000	40,000	0.56	0.26
40,000	46,000	0.53	0.26
46,000	70,000	0 (no credit)	0.26
70,000	80,000	0 (no credit)	0.25
80,000	86,400	0 (no credit)	0.23
86,400	2,592,000	0 (no credit)	0 (no credit)

Table 2
Time Constants for Containment Spray Removal of Particulate Fission Product
(One Train of Spray with Minimum Safeguards Failure)

Time Span (sec)		Time Constant (hr^{-1})	
		Baseline	Proposed
Start	End	CLB	Modifications
0	600	0.00	0.00
600	3,000	20.00	0.00
3,000	3,240	9.36	0.00
3,240	3,500	7.19	0.00
3,500	4,000	16.50	0.00
4,000	4,500	16.50	0.00
4,500	4,800	16.50	0.00
4,800	5,000	16.50	9.36
5,000	7,100	16.50	9.36
7,100	8,000	16.50	9.36
8,000	24,600	1.65	0.94
24,600	30,000	1.65	0.94
30,000	40,000	1.65	0.94
40,000	46,000	1.65	0.94
46,000	70,000	1.65	0.94
70,000	80,000	1.65	0.94
80,000	86,400	1.65	0.94
86,400	2,592,000	0 (no credit)	0 (no credit)

Table 3
Limiting Iodine Partition Fractions for Post-LOCA ESF Leakage in the Auxiliary Building

Iodine Partition Fraction

Baseline Values

Time Step End (hours)	In the ESF Pump Rooms	Outside the ESF Pump Rooms: Offsite	Outside the ESF Pump Rooms: Control Room
3	0.100	0.0134	0.013
72	0.028	0.0037	0.010
720	0.010	0.0013	0.010

Values with Proposed Modifications in Place

Time Step End (hours)	In the ESF Pump Rooms	Outside the ESF Pump Rooms: Offsite	Outside the ESF Pump Rooms: Control Room
3	0.100	0.0133	0.013
72	0.031	0.0041	0.010
720	0.010	0.0013	0.010

Table 4
Iodine Partition Fractions for Post-LOCA RWST Release Model for 20 GPM ECCS Backleakage (Proposed Modifications in Place)

Time-steps		Current Licensing Basis		Proposed Amendment	
Start Time (sec)	End Time (sec)	IODEX Release Fraction	Release Rate (cfm)	IODEX Release Fraction	Release Rate (cfm)
0	790	0.000	0.000	0.000	0.000
790	810	9.19×10^{-11}	2.459×10^{-10}	0.000	0.000
810	900	2.894×10^{-09}	7.739×10^{-09}	0.000	0.000
900	1200	3.443×10^{-08}	9.207×10^{-08}	0.000	0.000
1200	1400	9.799×10^{-08}	2.620×10^{-07}	0.000	0.000
1400	1800	1.772×10^{-07}	4.738×10^{-07}	0.000	0.000
1800	2160	3.486×10^{-07}	9.321×10^{-07}	0.000	0.000
2160	3600	3.486×10^{-07}	9.321×10^{-07}	1.049×10^{-06}	2.805×10^{-06}
3600	4800	4.228×10^{-07}	1.131×10^{-06}	1.111×10^{-06}	2.970×10^{-06}
4800	6000	4.128×10^{-07}	1.104×10^{-06}	1.044×10^{-06}	2.791×10^{-06}
6000	7200	3.916×10^{-07}	1.047×10^{-06}	9.723×10^{-07}	2.600×10^{-06}
7200	28,800	3.376×10^{-07}	9.027×10^{-07}	1.099×10^{-06}	2.938×10^{-06}
28,800	36,000	3.284×10^{-07}	8.781×10^{-07}	1.586×10^{-06}	4.240×10^{-06}
36,000	86,400	1.873×10^{-07}	5.008×10^{-07}	1.179×10^{-06}	3.152×10^{-06}
86,400	345,600	3.444×10^{-07}	9.209×10^{-07}	8.273×10^{-06}	2.212×10^{-05}
345,600	2,592,000	6.388×10^{-06}	1.708×10^{-05}	1.230×10^{-05}	3.289×10^{-05}

Table 5
Comparison of Radiological Consequences Results
(Rem TEDE)
(Proposed Modifications vs Baseline)

	Baseline Case			**Proposed Amendment**		
	EAB	**LPZ**	**Control Room**	**EAB**	**LPZ**	**Control Room**
Containment Leakage	8.15	1.67	2.19	10.92	1.98	2.7
ECCS Leakage	1.31	0.23	0.8	1.33	0.25	0.9
Total Dose (not airborne)	9.46	1.9	2.99	12.25	2.23	3.6
Direct Shine	N/A	N/A	1.26	N/A	N/A	1.26
Total Dose	**9.46**	**1.9**	**4.25**	**12.25**	**2.23**	**4.86**
Regulatory Limit	25	25	5	25	25	5

September 12, 2011

Mr. Regis T. Repko
Vice President
McGuire Nuclear Station
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12700 Hagers Ferry Road
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SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATION CHANGES TO ALLOW MANUAL OPERATION OF THE CONTAINMENT SPRAY SYSTEM (TAC NOS. ME4051 AND ME4052)

Dear Mr. Repko:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 265 to Renewed Facility Operating License NPF-9 and Amendment No. 245 to Renewed Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 28, 2010, as supplemented by letters dated November 15, 2010, March 23, 2011, and May 2, 2011.

The amendments revise the TSs to allow manual operation of the containment spray system and to change the setpoints for the refueling water storage tank.

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 301-415-1119.

Sincerely,
/RA/
Jon Thompson, 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. 265 to NPF-9
2. Amendment No. 245 to NPF-17
3. Safety Evaluation

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DATE	05/16/11	05/16/11	05/16/11	09/02/10	12/21/10	02/11/11	03/29/11	05/13/11
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