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

August 9, 2013

Mr. K. Henderson
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATIONS CHANGES FOR NUCLEAR SERVICE WATER SYSTEM (TAC NOS. ME7659 AND ME7660)

Dear Mr. Henderson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 271 to Renewed Facility Operating License NPF-35 and Amendment No. 267 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 22, 2011, as supplemented by letters dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013.

The amendments revise the Technical Specifications to allow single discharge header operation of the nuclear service water system for a time period of 14 days.

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

Sincerely,

Jason C. Paige, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 271 to NPF-35
2. Amendment No. 267 to NPF-52
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 271
Renewed License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Carolinas, LLC, acting for itself, and North Carolina Electric Membership Corporation (licensees), dated November 22, 2011, as supplemented by letters dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013, 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-35 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 271, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 9, 2013

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 267
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013, 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-52 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 267, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 9, 2013

ATTACHMENT TO

LICENSE AMENDMENT NO. 271

RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

LICENSE AMENDMENT NO. 267

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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

Remove	Insert
License Pages	License Pages
NPF-35, page 4	NPF-35, page 4
NPF-52, page 4	NPF-52, page 4
TS Pages	TS Pages
3.7.8-1	3.7.8-1
3.7.8-2	3.7.8-2
3.7.8-3	3.7.8-3
-	3.7.8-4

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 271 ,which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

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

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

(4) Antitrust Conditions

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 267 ,which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

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

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

(4) Antitrust Conditions

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

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

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

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

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

3.7 PLANT SYSTEMS

3.7.8 Nuclear Service Water System (NSWS)

LCO 3.7.8 Two NSWS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Not applicable while in Condition C of this LCO unless entry is directed by Note 2 of Condition C. ------------------------------ One NSWS train inoperable.	A.1 ------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by NSWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for residual heat removal loops made inoperable by NSWS. ------------------------------ Restore NSWS train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ------------NOTES------------ 1. Entry into this Condition shall only be allowed for pre-planned activities as described in the Bases of this Specification. 2. Immediately enter Condition A of this LCO if one or more NSWS components become inoperable while in this Condition and one NSWS train remains OPERABLE. 3. Immediately enter LCO 3.0.3 if one or more NSWS components become inoperable while in this Condition and no NSWS train remains OPERABLE. ------------------------------ One NSWS supply header inoperable due to NSWS being aligned for single supply header operation.	B.1 Restore NSWS supply header to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTES---------- 1. Entry into this Condition shall only be allowed for Unit 1 and for pre-planned activities as described in the Bases of this Specification. Entry into this Condition shall not be allowed while Unit 2 is in MODE 1, 2, 3, or 4. 2. Immediately enter Condition A of this LCO if one or more Unit 1 required NSWS components become inoperable while in this Condition and one NSWS train remains OPERABLE. 3. Immediately enter LCO 3.0.3 if one or more Unit 1 required NSWS components become inoperable while in this Condition and no NSWS train remains OPERABLE. ------------------------------ One NSWS train inoperable due to NSWS being aligned for single Auxiliary Building discharge header operation.	C.1 Restore NSWS train to OPERABLE status.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	-----NOTE----- Isolation of NSWS flow to individual components does not render the NSWS inoperable. ---------- Verify each NSWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	-----NOTE----- Not required to be met for valves that are maintained in position to support NSWS single supply or discharge header operation. ---------- Verify each NSWS automatic 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 3.7.8.3	Verify each NSWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 271 TO RENEWED FACILITY OPERATING LICENSE NPF-35

AND

AMENDMENT NO. 267 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By application dated November 22, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11327A149), as supplemented by letters dated July 9, 2012 (ADAMS Accession No. ML12194A218), November 12, 2012 (ADAMS Accession No. ML12319A075), January 28, 2013, (ADAMS Accession No. ML13032A006), and May 15, 2013, (ADAMS Accession No. ML13140A012), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2). The supplements dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published the *Federal Register* on May 15, 2012 (77 FR 28630).

The proposed changes would revise the TSs to allow single discharge header operation of the nuclear service water system for a time period of 14 days.

2.0 REGULATORY EVALUATION

The Nuclear Service Water System (NSWS) consists of two loops (A & B) of essential equipment, each of which is shared between units. Each loop supplies two trains of NSWS, one train for Unit 1 and one train for Unit 2. One train of NSWS is sufficient to perform the safety functions of NSWS for the applicable unit, thus each unit has redundant trains of NSWS. Two bodies of water serve as the Ultimate Heat Sink (UHS) supply to the NSWS. Lake Wylie is the normal source of NSWS and the Standby Nuclear Service Water Pond (SNSWP) is the emergency source. During normal operation the NSWS loops are cross connected and discharge in a single discharge header to Lake Wylie. Upon a loss of Lake Wylie, the supply to the NSWS pumps shift to the SNSWP, the common discharge to Lake Wylie is isolated, and the discharge of the A & B loops are isolated from each other and discharge separately to the SNSWP. Thus during a design basis accident, each unit has two trains of NSWS each having an independent and redundant supply and return. Each train is capable of performing the NSWS safety functions for that Unit, which satisfies the single failure criterion of General Design Criterion (GDC) 44 of 10 CFR Part 50, Appendix A.

Technical Specification (TS) 3.7.8 requires two trains of NSWS to be OPERABLE in order to meet the Limiting Condition for Operation (LCO). The TS Bases states that two NSWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power. Each train is OPERABLE when both NSWS pumps on the associated NSWS loop are OPERABLE and the associated piping, valves, and instrumentation and controls to perform the safety-related function are OPERABLE. If one train becomes inoperable, the required action is to restore the NSWS train to OPERABLE within 72 hours or shutdown to MODE 5. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a Design Basis Accident (DBA) occurring during this time period.

The licensee has proposed new Condition C for TS 3.7.8, which would allow one train of NSWS to be inoperable (but available through the other loops discharge path) for 14 days, when the cause of inoperability is due to the associated discharge of the inoperable train being discharged to the redundant train's discharge header.

New Condition C would allow planned maintenance on the Auxiliary Building NSWS discharge headers either between 1RPN20 and 1RN58B or between 1RPN19 and 1RN63A. Unit 1 would be in Mode 1, 2, 3, or 4 and Unit 2 in neither Modes 1, 2, 3, nor 4. Entry into this Condition is not allowed in response to unplanned events or for other events involving the NSWS. Examples of situations for which entry into this Condition is prohibited are emergent repair of discovered piping leaks and other component failures. For unplanned events or other events involving the NSWS, Condition A must be entered. The lineup required for new Condition C aligns the NSWS as follows: a) isolate the common discharge to Lake Wylie by shutting 1RN843B and 1RN57A; b) isolate either the B NSWS loop discharge by shutting 1RNP20 and 1RNP58B or the A NSWS loop discharge by shutting 1RNP19 and 1RNP63A, as applicable; c) open the crossover valves, 1RN53B and 1RN54A, to allow both A&B NSWS loops to discharge to the SNSWP through the A or B loop, whichever is not undergoing repair; d) remove power to crossover valves 1RN53B and 1RN54A and discharge isolation valves 1RN63A and 1RN58B, e) shut the discharge of the Containment Spray Heat Exchanger (either 1RN148A or 1RN229B as applicable) and remove power.

The Nuclear Regulatory Commission (NRC) requirements, review guidelines, and licensing basis that the staff considered applicable to the License Amendment Request (LAR) include:

> General Design Criterion 44, "Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection and isolation capabilities shall be provided to assure that for onsite electric

power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

General Design Criterion 5 - Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 1), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Reference 2), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in completion times (CTs). This regulatory guide also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensees' evaluation of the risk associated with a proposed CT TS change, as discussed below.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. (Key Principle 4)

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk significant operational factors, such as concurrent system or equipment testing, are also involved.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Tier 3 guidance can be satisfied by the Maintenance Rule 10 CFR 50.65(a)(4), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's CRMP and PRA model for this application.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 3), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light water-reactors.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 4). Guidance on evaluating PRA technical adequacy is provided in Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 5). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," (Reference 6), which includes CT changes as part of risk-informed decision making. Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

The Updated Final Safety Analysis Report (UFSAR) Section 9.2.1, "Nuclear Service Water System," provides the design basis and description of the system. The Nuclear Service Water System (NSWS) provides essential auxiliary support functions to Engineered Safety Features of the station. The system is designed to supply cooling water to various heat loads in both the safety and non-safety portions of each unit. Provisions are made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under accident conditions. Sufficient redundancy of piping and components is provided to ensure that cooling is maintained to essential loads at all times.

The Nuclear Service Water System is designed to withstand a safe shutdown earthquake and to prevent any single failure from limiting the ability for the engineered safety features to perform their safety functions. Sufficient pump capacity is included to provide the cooling water to shutdown each unit and the valves are arranged in such a way that loss of one train does not jeopardize the entire system. The RN System is designed to supply the cooling water requirements of a simultaneous LOCA on one unit and cooldown on the other unit assuming a single failure anywhere on the system, loss of offsite power and loss of Lake Wylie. Upon complete channel separation, both units are assured of having a source of water, at least one pump capable of supplying required flow on its associated channel, and at least one essential header to provide cooling water to components served by RN.

3.0 TECHNICAL EVALUATION

The NRC staff used RGs 1.174 and 1.177 in performing a detailed review of the licensee's request and compared the request against applicable regulatory criteria. This safety evaluation addresses PRA considerations and the other issues related to the risk-informed evaluations used by the licensee to support the LAR. The LAR and supplements provided all the risk-informed information required to support the three tiered approach described in RG 1.177. The NRC staff has organized the information into the three tier format to simplify review against RG 1.177 and support the risk-informed conclusions needed to determine the acceptability of the LAR. In completing this evaluation, the NRC staff considered the information that was provided by the licensee's LAR dated November 22, 2011, as supplemented by letters dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013.

3.1 Comparison Against Regulatory Criteria/Guidelines

The NRC staff's evaluation of the licensee's proposed changes using the three-tiered approach and the five key principles outlined in RGs 1.174 and 1.177, are presented in the following sections.

3.1.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 of the NRC staff's philosophy of risk-informed decision making, which concerns compliance with current regulations, evaluation of defense-in-depth, evaluation of safety margins, and performance monitoring strategies.

Key Principle 1: Compliance With Current Regulations

The licensee does not propose to deviate from existing regulatory requirements and compliance with existing regulations is maintained by the proposed LAR.

Key Principle 2: Evaluations of Defense-in-Depth

The elements of the defense-in-depth philosophy are described in RG 1.177. Consistency with defense in depth philosophy is maintained if:

- A reasonable balance among prevention of core damage, prevention of containment failure and consequence mitigation is preserved.

In the proposed lineup, the discharge header of one Unit 1 NSWS train will be realigned to the discharge header of the other Unit 1 NSWS train. This lineup renders the realigned NSWS train inoperable because it lacks complete independence and redundancy. However, the realigned NSWS train with all components operable other than its unique discharge header, will still provide cooling to its respective equipment that is important to safety, allowing the equipment to perform its safety functions through use of the common discharge header. Thus the realigned NSWS train will be inoperable, but available. In the event of a DBA, one operable Unit 1 train of NSWS and one inoperable/available Unit 1 train of NSWS are available to supply the Emergency Core Cooling Systems (ECCS), emergency diesel generators, and containment heat removal systems which prevent core damage and containment failure. Unit 2 will be required to be in either Mode 5, 6 or defueled, since one train of Unit 2 NSWS cooling loads will be inoperable and isolated.

This LAR does not affect liquid or gaseous radioactive effluents or filtration systems; therefore, radiological mitigation is not affected.

Based on the above described NSWS capability and the absence of any effect on radiological mitigation factors, the staff considers that a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved during the single discharge header alignment during a 14-day completion time.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

Programmatic activities to be used in accomplishing the proposed maintenance lineup include additional training on the NSWS single discharge header alignment in addition to the normal activities for preplanned maintenance. The staff does not consider this to be an over-reliance on programmatic activities and is reasonable for extending the Completion Time from 3 days to 14 days for the infrequent maintenance activities associated with this LAR.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency of challenges to the system.

When in new Condition C the discharge headers of the 1A and 1B NSWS trains are combined into one discharge header to the standby nuclear service water pond (SNSWP), which renders the 1A and 1B discharge headers neither independent nor redundant. This condition is a temporary relaxation of the redundancy requirements of GDC 44. A failure of the single common discharge header during a loss of coolant accident (LOCA) would cause the loss of both Unit 1 trains of NSWS. However, the failure of the single common discharge header is unlikely, considering the following lineup and precautions:

A) The single common discharge pipe is a moderate energy pipe whose pipe failure as the single failure after a DBA is not credible.
B) The motor operated valves (1RN63A or 1RN58B) in the single common discharge pipe will be opened with power removed to prevent inadvertent closure.
C) The manual valves (1RPN19 or 1RPN20) in the single common discharge pipe will be locked open and safety tagged.
D) The motor-operated crossover valves 1RN53B and 1RN54A will be tagged open with power removed to prevent these valves from repositioning and blocking flow.
E) The NSWS 1A and 1B supply headers will remain independent and redundant.
F) NSWS suction will be aligned to the SNSWP.

G) If any required NSWS component becomes inoperable and one NSWS train remains operable, the allowed completion time will revert back to 72 hours, otherwise an immediate entry into TS 3.0.3 is required.

Thus, the NRC staff considers that the single common discharge header lineup as proposed by the licensee has sufficient redundancy, independence, and diversity as a temporary lineup for the proposed completion time of 14 days.

- Defenses against potential Common-Cause Failures (CCFs) are maintained and the potential for introduction of new CCF mechanisms is assessed.

Possible CCFs are 1) breakage of the common discharge line, 2) failure of a Motor Operated Valve (MOV) or manual valve , 3) operator error (shutting an MOV or manual valve), and 4) valve failure or pipe breakage causing flooding.

As discussed above, breakage of the single common discharge line is not considered credible as a failure after a DBA-LOCA. However, breakage/leakage or flow blockage as an initiating event is a credible failure resulting in loss of NSWS. The staff was concerned that the licensee did not address these failures in the single common discharge header that would cause the loss of both NSWS trains. Therefore, the staff asked the licensee in the Requests for Additional Information (RAIs) dated May 11, 2012 and October 9, 2012, to discuss contingency plans training, procedure, and compensatory measures to restore NSWS in the event that the common discharge breaks/cracks or becomes flow blocked. In its responses dated July 9, 2012 and November 12, 2012, the licensee stated that discharge flow to Lake Wylie can be restored from the control room by repositioning MOVs 1RN57A and 1RN843B. The licensee also stated that the suction to the NSWS will also be realigned to Lake Wylie. The licensee stated that operators are trained for loss of NSWS and will receive additional training on the NSWS single discharge header alignment prior to implementation. The staff considers this response satisfactory due to the ability to restore flow by realigning to Lake Wylie.

Flow blockage by failure of a MOV could cause loss of NSWS or partial loss of NSWS. The licensee will open and remove power from the crossover MOVs (1RN53B and 1RN54A) and SNSWP isolation valve (1RN58B or 1RN63A as applicable). Removing power from these MOVs will eliminate spurious operation and operator error in unintended positioning of these MOVs. Manual valves 1RNP19 and 1RNP20 which either isolate the work area or are in line with the common discharge header will be positioned, safety tagged, and locked to prevent inadvertent operator action.

The NRC staff was concerned that failure of the work isolation boundaries, i.e. MOVs 1RN63A and 1RN58B or manual valves 1RNP19 and 1RNP20, could allow flooding in the Auxiliary Building or block NSWS flow. Therefore, in an RAI dated October 9, 2012, the staff asked the licensee to discuss the effects of complete valve failure or operator error that breaks a work boundary. In their response dated November 12, 2012, the licensee stated that credible failures could occur in the stem/disc pins or in an operator gear box while the valve is being positioned, but would not fail after positioning. However, these types of failure would be discovered before the piping is released for work. To preclude this failure mode, the valve disc pins will be verified to be installed, or, if not verifiable, a mechanical gagging device will be installed. Failure after the valve is closed is not credible because of the design features of these types of butterfly valves. Operator error would be averted by safety tagging the boundary valves, removing power from the MOVs, and locking the manual valves.

To address the staffs concern about siphoning/draining the SNSWP to the open work boundaries, the licensee responded in a letter dated July 9, 2012, that the normal SNSWP top of pond elevation is about 571ft. Mean Sea Level (MSL). The centerline of the work boundary pipe in the Auxiliary Building is about 581.25 ft. MSL. Therefore it would not be possible to siphon the SNSWP back into the Auxiliary Building if a boundary valve failed and there was not a simultaneous Probable Maximum Precipitation (PMP) event. A PMP event would only occur due to a major weather event, and the NSWS single discharge header alignment will have procedural requirements to verify that no severe weather is in the forecast prior to implementation. In the unlikely event of valve failure, the total amount of water released could be 170,000 gallons. The equipment in the affected spaces in the Auxiliary Building have been previously analyzed and qualified for the consequences of pipe rupture. If the entire 170,000 gallons were to drain to the 522 ft. MSL, it would drain to the Residual Heat Removal (RHR) Containment Spray Sump which has a capacity of approximately 218,000 gallons before the nuclear safety-related RHR minimum flow instrumentation could be impacted. If the flooding water entered the Auxiliary Feedwater System (AFW) sump via the floor drains, the operators would have 90 minutes to take action per the abnormal Operating Procedure for flooding. Operators are trained on existing procedures for plant flooding and loss of the NSWS. Operators will receive additional training on the NSWS single discharge header alignment prior to implementation.

Based on the NSWS capability and the protection from CCF described above, the staff considers that defenses against CCF potential are maintained during the single discharge header alignment for a 14-day completion time.

- Independence of physical barriers is not degraded.

The proposed TS change affects neither the fuel cladding, nor the reactor coolant pressure boundary, nor the containment. Thus, the independence of these barriers is not affected by the TS amendment.

- Defenses against human errors are maintained.

Possible human errors would include incorrect operation of valves. However, MOVs in the discharge headers that either isolate the work area or are in line to the common discharge header will be prepositioned and have power removed. Manual valves that isolate the work area will be shut and locked and tagged to prevent inadvertent operation. Personnel will be working to approved procedures and will be pre-trained for the evolution. Based on the above, the staff considers that defenses against human errors are maintained.

- The intent of the plant's design criteria is maintained.

The intent of GDC 44 is to require a cooling water system that meets the design function during a loss of offsite power and assuming a single failure. Any single failure in the common discharge line would prevent the cooling system from meeting its design function. However, as described above, the credible failures in the common discharge line are compensated by prepositioning valves, locking out power, using the safety tag system, and training operators. A temporary relaxation of redundancy and independence is recognized in Generic Letter 80-30, "Clarification of the Term "Operable" As It Applies to Single Failure Criterion for Safety Systems Required by TS," for approved TS conditions.

The licensee has performed calculations that show that when the NSWS is aligned in the Single Discharge Header alignment and the Train A discharge header to the SNSWP is out of service, the NSWS Train 1A components will receive sufficient flow. In this alignment, Train 1A NSWS discharge flow combines with the Train 1B and Train 2B NSWS discharge flow and discharges through the Train B header to the SNSWP. Likewise, the licensee has performed calculations that demonstrate that when the NSWS is aligned in the Single Discharge Header alignment and the Train B discharge header to the SNSWP is out of service, the NSWS Train 1B components will receive sufficient flow. In this alignment, Train 1 B NSWS discharge flow combines with the Train 1A and Train 2A NSWS discharge flow and discharges through the Train A header to the SNSWP. Based on the above, the staff considers that the intent of the plant's design criteria is maintained.

Key Principle 3: Evaluation of Safety Margins

The extended CT is not in conflict with Codes and Standards approved for use by the NRC relevant to the NSWS. Safety analysis acceptance criteria as specified in the UFSAR, particularly for the LOCA, are met during the extended CT, assuming no additional failures.

Key Principle 5: Performance Measurement Strategies - Implementation and Monitoring Program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of systems, structures, and components (SSCs) impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule (10 CFR 50.65) can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. The results of the risk evaluation are presented in the PRA tables later in this document for preventive maintenance (PM) and compared to the acceptance guidelines of RG 1.174 and RG 1.177.

Preventive maintenance is defined as planned maintenance and not the direct result of equipment failure. The NSWS is covered by performance monitoring and reliability goals via the Maintenance Rule (10 CFR 50.65(a)(4)); therefore it is assumed that the plant risk is minimized consistent with the requirements of the Maintenance Rule. Consistent with the Maintenance Rule, during NSWS PM activities, it is assumed that common cause failure contributors that affect both NSWS loops are not applicable and normal risk management measures are implemented, including that the unaffected NSWS loop is available. These measures also minimize the testing and maintenance (T&M) activities on other risk significant plant equipment. Specifically, it was assumed that no T&M on specific equipment that affects the reliability of the train associated with the operable NSWS train will be scheduled during the NSWS train out-of-service time.

3.2 PRA Technical Evaluation

The evaluation presented below addresses the NRC staff's assessment of the three tiered approach.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. (Key Principle 4)

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk significant operational factors, such as concurrent system or equipment testing, are also involved.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. Tier 3 guidance can be satisfied by the Maintenance Rule 10 CFR 50.65(a)(4), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application.

3.2.1 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed CT extension on plant operational risk based on the Catawba Probabilistic Risk Assessment (PRA) model. The Tier 1 staff review involves two aspects: (1) evaluation of the validity of the PRA and its application to the proposed CT extension, and (2) evaluation of the PRA used to support this application.

3.2.1.1 PRA Quality

To determine whether the PRA used in support of the proposed CT extension is of sufficient quality, scope, and detail, the staff evaluated the relevant PRA information provided by the licensee in its submittal, as supplemented, and considered the findings of recent PRA peer reviews and evaluations. The staff's review of the licensee's submittal focused on the capability of the licensee's PRA model to analyze the risks resulting from the proposed NSWS CT extension and did not involve an in-depth review of the licensee's PRA.

As stated in the submittal, the licensee is proposing to extend the NSWS TS CT from 72 hours to 14 days for single discharge header operation. Duke used the Catawba PRA model to evaluate the quantitative impacts of the TS change. This model is a full-power internal event, seismic, and internal fire risk model. As discussed above, the staff determined that the PRA quality is of sufficient scope and quality.

3.2.1.1.1 Internal Events

According to their response to the staff's RAI, the Catawba PRA initially received an internal events full scope peer review by an industry team in March 2002. There were seventy-five findings from this review where the internal events model was found not to conform to capability category II of the standard for certain supporting requirements (SRs). Of these, nine were related to this LAR. The licensee identified and dispositioned these findings for this application and the NRC staff reviewed their assessment as discussed below:

Finding 1 SR DA-B1 (open): Data calculations did not reflect segregation of standby and operating component data. The licensee stated that this is a refinement to the equipment failure rates and most components are grouped appropriately. It is conservative to not group components for parameter estimation according to characteristic, and failure to segregate the data has a conservative impact on calculated risk.

Finding 2 SR HR-A2 (open): Calibration activities that, if performed incorrectly, could have an adverse impact on the automatic initiation of standby safety equipment were not identified. The data from preliminary Electric Power Research Institute (EPRI) Human Reliability Analysis (HRA) calculations resulted in no significant impact on this application due to single channel failure and multiple channels failure are expected to fall in 1E-3 and 1E-5 demands, respectively, which has no substantive impact on the results.

Finding 3 SR HR-A3 (open): Maintenance and calibration activities that could simultaneously affect equipment in different trains of a redundant system or a diverse system were not identified. This SR was evaluated using the EPRI HRA calculator for multiple channels which resulted in a 1E-5 failure range which has no substantive impact on the results.

Finding 4 SR HR-D6 (open): An assessment of the uncertainty in the human error probabilities were not included when providing point estimates. The licensee set the pre-initiator values high, which bound the mean values. This is a conservative treatment.

Finding 5 SR HR-G9 (open): Mean values for post-initiator HEPs were not included. The licensee set the post initiator values high which bound the mean values. This is a conservative treatment.

Finding 6 SR IF-C2c (open): The review identified one instance of inconsistency in the prior and posterior distributions for Bayesian updated data, and a lack of documentation of any evaluation of these distributions for consistency. The licensee performed these evaluations and adjusted the data to account for inconsistencies, resulting in a minor increase in calculated risk.

> Finding 7 SR IF-C3 (open): Identify the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. The licensee's evaluation of Internal/External Flood analysis confirms there is not a significant impact from flooding for this LAR.
>
> Finding 8 SR IF-C3b (open): Identify more flood propagation analysis through the normal flow path from one area to another via drain lines and areas connected via back flow through drain lines involving failed check valves, pipe, and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. The licensee's evaluation of Internal/External Flood analysis confirms there is not a significant impact from flooding for this LAR.
>
> Finding 9 SR IF-E6b (open): Address the indirect effects (e.g. submergence, jet impingement, and pipe whip) of the flood. The licensee's evaluation of Internal/External Flood analysis confirms there is not a significant impact from flooding for this LAR.

The staff reviewed the licensee's assessment of the nine findings and determined that it is consistent with RG 1.200.

In a letter dated July 9, 2012, the licensee stated that the Catawba PRA was updated in 2005 and included an upgrade to the large early release frequency (LERF) model. In a letter dated May 15, 2013, the licensee provided a focus-scoped peer reviewed analysis of the upgraded LERF model. The peer review resulted in two SRs being not met.

The focused scope peer review stated that Catawba used the conservative parameter estimates from NUREG/CR-6595, Rev. 0, to characterize the accident progression phenomena. RG 1.174 recognizes that NUREG/CR-6595 (Reference 7) may be used when the quantitative LERF results are not close to any acceptance guidelines. The licensee's LERF results used in the LAR are not near the acceptance guidelines. All findings and observations (F&O) noted the use of the older version of NUREG/CR-6595 and the licensee summarized its engineering analysis to defend the use of the older values. The licensee provided an engineering evaluation which showed their numbers to be comparable to the updated standard. Also, their estimated LERF values were an order of magnitude below the threshold for Region III, which is an acceptable alternative to a detailed analysis outlined in NUREG/CR-6595. The peer review team comments on Duke Energy's engineering analysis states that the analysis appears to have a reasonable basis for using the Rev. 0 CCFP values based on plant-specific analysis. The licensee provided its analysis in the disposition of the F&O. The NRC staff has reviewed this analysis and concurs with the engineering analysis to justify the use of the older version of NUREG/CR-6595.

3.2.1.1.2 Seismic Risk

The seismic PRA analysis results screened out the contribution of NSWS structures and components. A sensitivity study was performed to conservatively bound the impact of having an NSWS discharge line unavailable during a seismic event. For the seismic analysis, components with median seismic capacities in excess of 2g were screened out of the seismic fault tree models due to low probability of failure. Structures were also eliminated from consideration when their seismic capacities were in excess of 2.5g. The licensee considered the Auxiliary Feedwater Motor-Driven Pump 'B,' Safety Injection Train 'B,' Emergency Diesel Generator 'B,' Component Cooling Train 'B,' and Residual Heat Removal Train 'B' unavailable and the standby trains of these systems available for their cutsets. This is consistent with the previous individual plant examination (IPE) and individual plant examination of external events (IPEEE) seismic analysis where seismic capacities above 2g for components and 2.5g for structures were eliminated. The licensee's PRA results indicate that this is a conservative bounding value since both trains of each safety system would normally be available, due to this technical specification only applying to the inoperability of a single discharge header that is out of service. The NRC reviewed the licensee's seismic risk and determined that it is consistent with Catawba's current licensing basis.

3.2.1.1.3 Other Events

The licensee's IPE and IPEEE submittals included the same analysis and methodology as the current fire PRA. Internal fire events contributed approximately 10% of the CDF and 3% of the LERF. However, the contribution of fire-induced loss of NSWS events did not appear in any core damage or Large Early Release Sequences. Furthermore, the licensee's configuration of the NSWS is with the essential header discharge valves and the discharge valve to the SNSWP remaining open during the requested 14-day CT with power removed. Fire events affecting power to these valves would have no effect on the NSWS discharge alignment during the CT. Therefore, the licensee determined that the contribution of fires was deemed negligible, in which the staff agrees with this determination.

The licensee recently updated the flooding analysis. The internal/external flood evaluations did not find any vulnerability due to floods. In addition, the IPEEE results show that the licensee's evaluation of probable maximum flood resulting from probable maximum precipitation did not indicate any plant vulnerability. Therefore, the impact due to internal/external flooding is insignificant to this application.

Tornado/high winds hazards and postulated transportation accidents were screened out as not significant. The licensee evaluated occurrence frequency, tornado missile analysis, and tornado wind analysis, and concluded that one event (Tornado Causes a Loss of Offsite Power) contributed to the CDF (~3%) and the LERF (~4%). The staff reviewed the licensee's flooding analysis and tornado/winds evaluation and determined that it is acceptable.

3.2.1.2 PRA Insights

Based on the Catawba PRA model, the addition of risk management actions, and the availability of the NSWS when placed in the operating configuration through a single discharge header, the licensee calculated values for delta core damage frequencies (ΔCDF), incremental conditional core damage probability (ICCDP), delta large early release frequency (ΔLERF), and incremental conditional large early release probability (ICLERP) for the proposed 14-day NSWS CT. The ΔCDF and ΔLERF evaluation was performed assuming that an extended 14-day CT would be extended twice (once per train) each year although the extended CT would be entered on a decreasing duration and frequency after the initial year. The results of the risk evaluations are presented in the tables below for maintenance and compared to the acceptance guidelines of RG 1.174 and RG 1.177.

NSWS EXTENDED CT INTERNAL EVENTS PRA ΔCDF, ΔLERF, ICCDP, AND ICLERP RESULTS		
Risk Metric	Acceptance Guideline[1]	PRA Results/rx-yr
ΔCDF	< 1.0E-6/reactor-year	2.00E-07
ICCDP	< 5.0E-7	9.37E-08
ΔLERF	< 1.0E-7/reactor-year	1.08E-8
ICLERP	< 5.0E-8	3.84E-09
NSWS EXTENDED CT SEISMIC PRA ΔCDF, ΔLERF, ICCDP, AND ICLERP RESULTS		
Risk Metric	Risk Metric	Risk Metric
ΔCDF	< 1.0E-6/reactor-year	2.1E-6
ICCDP	< 5.0E-7	8.1E-8
ΔLERF	< 1.0E-7/reactor-year	NOT INCLUDED
ICLERP	< 5.0E-8	NOT INCLUDED

1 Acceptance guidelines for very small changes. Acceptance guidelines for small changes are an order of magnitude higher.

2 It is anticipated that Catawba will enter this TS Condition for the full 14 days twice (once per train) initially and then in subsequent refueling cycles enter the TS Condition on a decreasing duration and frequency. Therefore, the values for the subsequent refueling cycles will be less than the values reported in the delta.

The risk values in the Unavailability tables are within the RG 1.177 and RG 1.174 acceptance guidelines for a very small incremental increase in risk (i.e., ICCDP and CLERP) and a small increase in the change in risk (i.e., ΔCDF and ΔLERF). Based on RG 1.174 guidance for changes in CDF within the range of 1E-6/year to 1E-5/year, an application would be considered only if the total CDF is less than 1E-4/year. This is the case for Catawba as the delta CDF remains within the RG 1.174 acceptance guidelines when internal events PRA and seismic PRA results are considered.

The PRA scope addresses internal events, seismic events, and fires during full power operation. The Catawba plant PRA has been maintained to reflect Regulatory Guide 1.200 and 1.174 criteria in support of the NSWS CT extension request and to represent additional plant operating history and component failure data. The PRA model used to perform this risk evaluation took into account previous modifications that allow the station to operate all four NSWS pumps via a single train when a NSWS supply header is removed from service. Based on the above discussion, the staff concludes that Tier 1 (fourth key principle) of risk-informed decision making is satisfied by the licensee's proposed amendment.

3.2.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

A licensee must provide reasonable assurance that risk significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service in accordance with the proposed TS change. The avoidance of risk-significant plant configurations limits potentially high risk configurations that could exist if equipment, in addition to that associated with the proposed TS change, is simultaneously removed from service or other risk-significant operational factors such as concurrent system or equipment testing are involved. Therefore, Tier 2 helps ensure that appropriate restrictions are placed on dominant risk-significant configurations relevant to the proposed TS change.

The licensee's evaluation identified the following conditions associated with the proposed extended NSWS CT extension.

While the NSWS is aligned in the single Auxiliary Building discharge header configuration, procedures will direct compensatory measures. The procedures for aligning the NSWS in the single Auxiliary Building discharge header configuration will include the following measures:

- Verify flood barriers in the Turbine Building basement prior to implementation.
- Verify no severe weather is in the forecast prior to implementation.
- Unit discretionary maintenance in Unit 1 and Unit 2 cable rooms, on available diesel generators (DGs), on Unit 1 turbine-driven AFW pump, on plant Drinking Water System (which provides backup cooling to the Train A centrifugal charging pump), and on Standby Shutdown System (including standby makeup pump) during implementation.
- Implement roving fire watch in Unit 1 and Unit 2 cable rooms during implementation.

Based on its review of the licensee's evaluation, the staff has determined that Tier 2, Avoidance of Risk-Significant Plant Configurations is consistent with RG 1.177.

3.2.3 Tier 3: Risk-Informed Configuration Risk Management

A Tier 3 program ensures that while a NSWS is in a LCO condition, additional activities will not be performed that could further degrade the capability of the plant to respond to a condition the inoperable NSWS was designed to mitigate, and as a result, increase plant risk beyond that assumed by the risk-informed licensing action. Tier 3 programs: (1) ensure that additional maintenance does not increase the likelihood of an initiating event intended to be mitigated by the out-of-service equipment, (2) evaluates the effects of additional equipment out-of-service during NSWS maintenance activities that would adversely impact NSWS CT risk such as from redundant or associated systems or components, and (3) evaluates the impact of maintenance on equipment or systems assumed to remain operable by the NSWS CT analysis.

Accordingly, a licensee should develop a CRMP to ensure that it appropriately evaluates the risk impact of out-of-service equipment before performing a maintenance activity. Licensees can utilize the overall CRMP (as referenced in RG 1.177) through the Maintenance Rule (10 CFR 50.65(a)(4)). Specifically, the rule requires that, before performing any maintenance activity, the licensee must assess and manage the potential risk increase that may result from a proposed maintenance activity. A licensee's submittal must include a discussion of its CRMP for assessing the risk associated with the equipment affected by its application (i.e. NSWS single discharge header) and their conformance to the requirements of the Maintenance Rule, and the additions and clarifications outlined in Section 2.3.7.2 of RG 1.177, as they relate to this application.

The licensee has a developed a CRMP based on the Maintenance Rule and the staff has reviewed this CRMP. This program is a procedure-based, risk-informed assessment process to manage the risk associated with planned and unplanned (emergent) plant maintenance activities. Catawba PRA Workplace Procedure XSAA-106, "Workplace Procedure for PRA

Maintenance, Update and Application," controls this process, which also assesses the requirement for permanent plant changes that have an impact on the PRA model but have not been incorporated. The CRMP uses an integrated approach of both quantitative and qualitative methods to identify risk-significant plant maintenance equipment outage configurations. The CRMP performs a configuration-dependent assessment of the overall impact on risk of proposed plant configurations prior to, and during, the performance of maintenance activities that remove equipment from service. The program evaluates defense-in-depth of key plant safety functions associated with the maintenance activity. In addition, the licensee's Tier 2 commitments specify additional compensatory measures to assess and manage the risk for an extended NSWS CT.

Based on the discussion above, the staff finds the licensee's Tier 3 program for complying with the Maintenance Rule, is consistent with the guidance of Chapter 16.1 of the SRP and RG 1.177, and thus is acceptable.

3.3 Summary

The licensee provided all the information required to support a risk-informed decision on a proposed TS change as described in RG 1.177. The risk impact of the proposed extended NSWS single discharge header operation 14-day CT, as estimated by ΔCDF, ΔLERF, ICCDP, and ICLERP, is consistent with the acceptance guidelines specified in RG 1.174, RG 1.177, and staff guidance outlined in Chapter 19.0, Section 16.1 of NUREG-0800. The staff finds that the PRA and risk analysis approach used by the licensee to estimate the risk impacts were reasonable and of sufficient quality for the proposed amendment request. The licensee identified a risk-significant plant equipment configuration requiring TS, procedure, or compensatory measures. This was identified by the licensee and included as a licensing commitment. Based on the staff's review of the licensee's risk-informed assessment as discussed in this safety evaluation, the staff finds that the proposed extension of the NSWS Single Discharge Header Operation CT to 14 days at Catawba is acceptable based on the fact that the increase in plant risk is small and consistent with the acceptance guidelines of RG 1.177 and RG 1.174.

The safety-related functions of the NSWS system are maintained during the extended CT, although complete independence and redundancy in the discharge headers of the two trains of NSWS for Unit 1 are not maintained during the extended CT. As a result, one train of NSWS is considered inoperable but still available to provide cooling to safety-related loads. The plant would be vulnerable to a single failure in the common discharge line, but the credible single failures are eliminated by prepositioning MOVs and removing power, locking and safety tagging manual valves, and having procedures in place to restore cooling from Lake Wylie, and requiring Unit 2 to be shutdown and cooled down.

The requirements of GDC 44 are maintained, recognizing that a temporary relaxation of the single failure criteria in the common discharge header during the extended CT is acceptable. It is acceptable with the additional valve lineup requirements, additional operator training, and the ability to restore cooling from Lake Wylie if needed. With Unit 2 already cooled down, the requirements of GDC 5 are maintained in that the single discharge header lineup can accommodate both Unit 1 NSWS discharges and the remaining Unit 2 NSWS train.

The proposed license amendment request was evaluated by the NRC staff to determine whether applicable regulations and requirements continue to be met. The NRC staff determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TSs. Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, and sufficient safety margins will be maintained. The NRC staff, therefore, finds this license amendment request acceptable.

4.0 REGULATORY COMMITMENTS

Below is a list of commitments made by the licensee:

- Duke Energy commits to include corresponding detail regarding the single Auxiliary Building discharge header alignment in the UFSAR following NRC approval of this amendment request (Duke Energy Carolinas, LLC, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Proposed Technical Specifications and Bases Amendment, TS and Bases 3.7.8, Nuclear Service Water System, Response to NRC Request for Additional Information, July 9, 2012).

- Procedural controls to prevent and mitigate the effects of flooding as discussed in the responses to Questions 3 and 4 of the attachment (Duke Energy Carolinas, LLC, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Proposed Technical Specifications and Bases Amendment, TS and Bases 3.7.8, Nuclear Service Water System, Response to NRC Request for Additional Information, November 12, 2012).

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 (77 FR 28630). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

8.0 REFERENCES

1. USNRC, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision on Plant-Specific Changes to the Licensing Basis, Revision 2," May 2011.
2. USNRC, Regulatory Guide RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, Revision 1," May 2011
3. USNRC, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1," January 2007.
4. USNRC, Standard Review Plan Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007.
5. USNRC, Standard Review Plan Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 3," September 2012.
6. USNRC, Standard Review Plan Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications, Revision 1," March 2007.
7. W.T. Pratt et al., "An Approach for Estimating the Frequencies for Various Containment Failure Modes and Bypass Events," NUREG/CR-6595, Revision 1, October 2004

Principal Contributors: G. Purciarello, NRR
J. Evans, NRR

Date: August 9, 2013

August 9, 2013

Mr. K. Henderson
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATIONS CHANGES FOR NUCLEAR SERVICE WATER SYSTEM (TAC NOS. ME7659 AND ME7660)

Dear Mr. Henderson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 271 to Renewed Facility Operating License NPF-35 and Amendment No. 267 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 22, 2011, as supplemented by letters dated July 9, 2012, November 12, 2012, January 28, 2013, and May 15, 2013.

The amendments revise the Technical Specifications to allow single discharge header operation of the nuclear service water system for a time period of 14 days.

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

Sincerely,

/RA/

Jason C. Paige, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 271 to NPF-35
2. Amendment No. 267 to NPF-52
3. Safety Evaluation

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