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

February 13, 2009

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

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2, ISSUANCE OF AMENDMENT REGARDING ONE-TIME EXTENSION OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 3.3.1.4 FREQUENCY (TAC NO. ME0403)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 242 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Unit 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 20, 2009, as supplemented by letter dated February 5, 2009.

The amendment would allow a one-time limited duration extension of the TS Surveillance Requirement (SR) 3.3.1.4 frequency. SR 3.3.1.4 is a Trip Actuating Device Operational Test of the reactor trip breakers and reactor trip bypass breakers.

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

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

Sincerely,

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

Docket No. 50-414

Enclosures:
1. Amendment No. 242 to NPF-52
2. Safety Evaluation

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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. 242
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 January 20, 2009, as supplemented by letter dated February 5, 2009, 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. 242, which are attached hereto, are hereby incorporated into this license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie C

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

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

Date of Issuance: February 13, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 242

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 Page 4	License Page 4
TSs 3.3.1-10	TSs 3.3.1-10

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 242 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

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

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

(4) Antitrust Conditions

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (continued)

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

(continued)

* The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Unit 2 Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling Outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this extension shall expire. The provisions of SR 3.0.2 are not applicable to this extension.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 242 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

1.0 INTRODUCTION

By application dated January 20, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090260224), as supplemented by letter dated February 5, 2009, (ADAMS Accession No. ML090420025), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Unit 2 (Catawba 2). The supplement dated February 5, 2009, 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 in the *Federal Register* on January 28, 2009 (74 FR 4986).

The proposed change would allow a one-time limited duration extension of the TS Surveillance Requirement (SR) 3.3.1.4 frequency from 62 days to 82 days. SR 3.3.1.4 is a Trip Actuating Device Operational Test (TADOT) of the reactor trip breakers (RTBs) and reactor trip bypass breakers. The current SR 3.3.1.4 specifies TADOT on 62 days frequency on a staggered test basis. The proposed amendment would add the following note to it:

> The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Unit 2 Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009 at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this extension shall expire. The provisions of SR 3.02 are not applicable to this extension.

The proposed change would allow deferral of a planned repair of the train 2A reactor trip bypass breaker cubicle cell switch, located in a hazardous area, until an upcoming refueling outage.

2.0 REGULATORY EVALUATION

The NRC staff used the following regulatory bases and guidance in its evaluation of the license amendment request:

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes the fundamental regulatory requirements. Specifically, 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," addresses the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

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

Section 50.55a(h)(2) requires that protection systems be consistent with their licensing basis of the Institute of Electrical and Electronics Engineers (IEEE) 603-1991 for plants with construction permits issued before January 1, 1971, or that protection systems meet IEEE 279-1971 or IEEE 603-1991 for plants with construction permits issued after January 1, 1971, but before May 13, 1999. Section 4.2 of IEEE 279-1971 discusses the general functional requirement for protection systems to ensure that they satisfy the single failure criterion.

Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (known an the Maintenance Rule), requires licensees to monitor the performance or condition of systems, structures, and components (SSCs) against licensee established goals in a manner sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions. The implementation and monitoring program guidance of Section 2.3 of Regulatory Guide (RG) 1.174, Revision 1, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued November 2002, and Section 3 of RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued August 1998, states that monitoring performed in conformance with the Maintenance Rule can be used when it is sufficient for the SSCs affected by the risk-informed application. In addition, 10 CFR 50.65(a)(4), as it relates to the proposed surveillance, bypass times, and CTs, requires the assessment and management of the increase in risk that may result from the proposed maintenance activity.

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 (Ref. 5). Guidance on evaluating Probabilistic Risk Assessment (PRA) technical adequacy is provided in Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 6). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decision Making: Technical Specifications," (Ref. 7).

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 or rule change.
- 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 NRC staff also used RGs to facilitate its review of the application. In particular, the NRC staff used the following RGs to guide its review:

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Ref. 2), 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 RG also provides risk acceptance guidelines for evaluating the results of such evaluations. While not directly applicable to temporary changes, the staff used this RG for guidance in evaluating the licensee's proposed changes.
- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," (Ref. 3), describes an acceptable risk-informed approach specifically for assessing proposed permanent TS changes in allowed outage times. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such assessments. While not directly applicable to temporary changes, the staff used this RG for guidance in evaluating the licensee's proposed changes.
- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Ref. 4), describes one 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.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's analysis in support of its proposed license amendment, which are described in the original submittal dated January 20, 2009, as supplemented by letter dated February 5, 2009.

3.1 Detailed Description of the Proposed Change

The proposed change would allow a one-time limited duration extension of the Technical Specification (TS) Surveillance (SR) 3.3.1.4 frequency from 62 days to 82 days. SR 3.3.1.4 is a Trip Actuating Device Operational Test (TADOT) of the reactor trip breakers (RTBs) and reactor trip bypass breakers. The current SR 3.3.1.4 specifies TADOT on 62 days frequency on a staggered test basis. The proposed amendment would add the following note to it:

> The SR 3.3.1.4 Frequency of "62 days on a STAGGERED TEST BASIS" as it applies to Unit 2 Train 2A and Train 2B reactor trip breaker testing may be extended on a one-time basis to March 10, 2009, at 0500 hours, upon which Unit 2 shall be in Mode 3 with reactor trip breakers open for the End of Cycle 16 Refueling outage. Upon entry into Mode 3 with reactor trip breakers open for this refueling outage, this extension shall expire. The provisions of SR 3.02 are not applicable to this extension.

The proposed change would allow deferral of a planned repair of the train 2A reactor trip bypass breaker cubicle cell switch, located in a hazardous area, until an upcoming refueling outage. The SR 3.3.1.4 is set to expire on February 19, 2009. Duke is requesting a one-time extension of the SR for Train 2A and 2B RTBs until March 10, 2009 at 0500 hours.

Normally, two series-connected RTBs, one for each reactor trip system train, deliver power from the rod control motor-generator sets to the rod control power cabinets. Each RTB has a reactor bypass breaker connected in parallel to it to facilitate on-line testing on one RTB at a time, without interrupting power to the rod drive mechanisms.

Under-voltage coils in RTBs keep the RTBs closed. A loss of power to any of these cabinets causes all rods to drop into the core. TS SR 3.3.1.4 is a TADOT of the RTBs and reactor trip bypass breakers. This SR must be performed on each bypass breaker prior to placing the associated RTB in service. The current SR 3.3.1.4 frequency is 62 days on a staggered test basis.

On January 8, 2009, the licensee experienced a problem in racking in Unit 2, Train 2A, reactor bypass breaker prior to the testing of the RTBs. As a result of the problem the licensee could not perform the scheduled testing of Train 2A RTB in time. During a conference call with the NRC staff on February 2, 2009, the licensee explained that it could not complete a root cause analysis of the problem, but that it expects the problem to be with the Train 2A reactor trip bypass breaker cubicle cell switch which the licensee cannot check while the unit is on-line. The proposed one-time extension will allow the TADOT of the Train 2A reactor trip bypass breaker cubicle cell switch during the next refueling outage and will thereby prevent any unnecessary trip of the associated RTB. Therefore, to avoid unnecessary shutdown of Unit 2, the licensee is requesting a one-time extension of this SR.

The licensee experienced similar cell switch problems during testing of RTB and bypass breakers on May 26, 1991, September 23, 2004, September 7, 2006, and November 16, 2006. However, none of these problems resulted in any breaker failure to trip open. As a consequence of these failures, the licensee has initiated scheduled preventative maintenance to replace the cell switches approximately every 7.5 years. The licensee also reviewed the test results of all RTB and reactor trip bypass breakers over the last 3 years and observed no problem between January 1, 2006, and December 31, 2008, i.e. no operational problems during the last 80 tests.

3.2 Key Information Used in the Review

The key information used in the NRC staff's review is contained in Attachment 4 and Section 3.0 of Attachment 1 of the application dated January 20, 2009 (Reference 1) and the response to an NRC staff request for additional information sent by letter dated February 5, 2009.

3.3 Comparison Against Regulatory Criteria/Guidelines

The NRC staff's evaluation of the licensee's proposed changes to the SR used the five key principles outlined in RGs 1.174 and 1.177 is presented in the following sections. The NRC staff also addressed the deterministic aspects of key principles 1, 2, 3 and 5 identified in SRP Section 19.2, and as listed in Section 3.3.1 of this SE, which include compliance with current regulations, evaluation of defense in depth, evaluation of safety margins, and monitor the impact of the proposed change using performance measurement strategies.

3.3.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3 and 5 of the 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 Principles 1 and 2: Compliance with Current Regulations and Defense in Depth

The proposed changes do not involve changes to instrument actuation setpoints, setpoint tolerance, testing acceptance criteria, or channel response times. No hardware changes are proposed or required to implement these changes at the plant. As required by 10 CFR 50.55a(h)(2), the protection systems must be consistent with the plant licensing basis or IEEE 603-1991 for plants with construction permits issued before January 1, 1971, or the protection systems must meet IEEE 279-1971 or IEEE 603-1991 for plants with construction permits issued after January 1, 1971, but before May 13, 1999.

The proposed SR frequency for RTB was recently revised by Amendments 247/240 for Catawba 1 and 2 from 31 days to 62 days on a staggered test basis. Amendments 247/240 were issued by letter dated December 30, 2008, for several RTS system and engineered safety feature actuation system (ESFAS) instrumentations. In its application dated January 20, 2009, the licensee stated that no change is required to the RTS or ESFAS instrumentation design such that compliance with any of the regulatory requirements and guidance documents listed in this SE would come into question and that the plant would continue to comply with applicable regulatory requirements.

Key Principle 3: Evaluation of Safety Margins

The proposed TS change is for a one-time extension of the RTB TADOT and does not involve any instrument drift or safety margins.

Key Principle 5: Monitor Impact of the Proposed Change Using Performance Measurement Strategies

As the proposed TS change is for a one-time extension of TADOT by approximately 20 days, monitoring of the proposed TS change is of limited scope.

The NRC staff concludes based on the information above that the proposed TS change complies with the deterministic aspects of the key principles 1, 2, 3 and 5.

3.3.2 Staff Technical Evaluation (PRA)

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making, that when the proposed changes result in a change in Core Damage Frequency (CDF) or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (Key Principle 4)

3.3.2.1 PRA Capability and Insights

The NRC staff reviewed (1) evaluation of the validity of the Catawba PRA model and their application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Quality

The objective of the PRA quality review is to determine whether the Catawba PRA used in evaluating the proposed changes is of sufficient scope, level of detail, and technical adequacy for this application. The NRC staff evaluated the PRA quality information provided by the licensee in their submittal, including industry peer review results.

The Catawba PRA model is a full scope PRA including both internal and external events. The PRA models those systems needed to estimate CDF and large early release frequency (LERF). This includes major support systems (e.g., ac power, service water, component cooling, and instrument air) as well as mitigating systems (e.g. emergency core cooling). The systems are generally modeled down to the component level including pumps, valves, and heat exchangers.

As stated by the licensee in a previous license amendment request dated December 30, 2008 (Reference 8), the following is a list of peer reviews conducted on the PRA modeling which assures the technical adequacy of the existing PRA:

- A peer review sponsored by the Electric Power Research Institute (EPRI) was conducted on the original Catawba PRA dated August 18, 1987.

- In March 2002, a peer review of the Catawba 1 and 2 PRA was conducted as part of the Westinghouse Owners Group PRA Certification Program.

The licensee did not identify any plant-specific design or operability issue that would invalidate the results. Based on NRC staff's review of peer review open items for the December 30, 2008, license amendment regarding RTB surveillance test interval extension (Reference 8), the NRC staff concludes none of the open items are expected to have a significant impact on PRA results.

In August 2008, a PRA Technical Adequacy Self-Assessment was conducted against the Supporting Requirements in the ASME standard (American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME-RA-Sc-2007) and RG 1.200 for Catawba 1 and 2.

The licensee identified that the PRA meets 224 of 306 ASME PRA Standard Supporting Requirements, as modified by Regulatory Guide 1.200. The licensee stated 24 of the Supporting Requirements are not applicable to the Catawba PRA, either because the referenced techniques are not utilized in the PRA or because the Supporting Requirement is not required by Capability Category II. Of the remaining 58 open Supporting Requirements, the licensee stated that 15 are of a technical nature. The NRC staff reviewed the 15 technical issues and determined that:

- Calibration of human error probabilities are not expected to contribute significantly to equipment unavailability.

- Initiating frequencies associated with fire and floods are not significant contributors to the risk impact for this amendment.

- Internal flood sequences which account for four of the technical issues are not significant contributors for this amendment.

- Loss of coolant accident and interfacing systems loss of coolant accident sequences have no significant impact for this amendment.

Based on review of the open technical issues, the NRC staff concludes that none of the open items are expected to have a significant impact on PRA results or insights.

The licensee identified a 5×10^{-10} truncation level used to generate the model cutsets used for risk analyses which supports this application. The licensee further identified sensitivity analyses it had performed on its baseline PRA model which confirms the adequacy of the selected truncation level, and also identified that the truncation level was consistent with the internal events PRA standard.

The licensee stated that the external events modeling (seismic, fire, and tornado) that exists in the Catawba PRA is at the level of detail used to support Individual Plant Examination External Events (IPEEE, Generic Letter 88-20) submittals and is consistent with the ASME standard and Regulatory Guide 1.200 supporting requirements. The licensee noted that RTS actuation signal failures and unavailability are very small contributors to the CDF for external events. This is consistent with the licensee's expectations because the dominant fire scenario involves failure of the Component Cooling Water System, and so the unavailability of the RTBs would not be relevant to such scenarios and therefore, fire risk is not a significant risk contributor to this application. Seismic considerations are negligible since other key plant equipment and supporting systems are more susceptible to the impact of a seismic event than the reactor vessel internals. Therefore, seismic risk is not a significant risk contributor to this application.

Based on review of the above information, the NRC staff finds that the licensee has satisfied the intent of RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), RG 1.174 (Section 2.2.3 and 2.5), and SRP Section 19.1, and that the quality of the Catawba internal events PRA is sufficient to support the risk evaluation for internal events provided by the licensee in support of the proposed license amendment.

PRA Results and Insights

The risk metrics for ΔCDF and ΔLERF for internal events were calculated by the licensee by using an increased failure probability for the basic event due to the surveillance test interval extension. The choice of the revised failure probability was made using an NRC-approved methodology discussed in the Industry Implementation Guidance for TSTF-358, Revision 6, "Missed Surveillance Requirements", TSTF-IG-06-01. Since the proposed extension is being requested for both Train 2A and Train 2B RTBs, the common cause event, common cause failure of Reactor Trip Breakers to Open, increased linearly from 1.6E-05 to 2.6E-05.

The licensee's methodology is consistent with the guidance of RG 1.177, Section 2.3.4, and is therefore acceptable to the NRC staff.

The results of the licensee's analyses of internal events (Table 1) meet Regulatory Guide 1.174 guidelines, and are therefore acceptable to the NRC staff.

Table 1: Internal Events Risk

Risk Metric	**Catawba Result**	**RG 1.174 Guidance – Very Small Changes**
ΔCDF	6.0E-08/year	1.0E-06/year
ΔLERF	3.2E-08/year	1.0E-07/year

In order to address potential uncertainties in the analysis results related to the increased failure probability of the RTBs due to the surveillance interval extension, the licensee conducted a sensitivity study of the above results by increasing the original failure probability by a factor of three. The results obtained from this evaluation were still well within Regulatory Guide 1.174 guidance, and adequately demonstrated that uncertainty considerations would not have any significant impact on the regulatory decision.

3.4 NRC Staff's Findings and Conditions

The NRC staff has reviewed the information provided by the licensee on past performance and considered the proposed TS change for a one-time extension of reactor trip breaker TADOT by

62 days to 82 days. The NRC staff concludes that the proposed TS change complies with the regulatory requirements specified in Section 2.0 or this safety evaluation and is therefore acceptable.

The risk impacts of extending SR 3.3.1.4 by 20 days are within the acceptance guidelines of RG 1.174 which are applicable for permanent changes to the licensing basis, and are therefore acceptable for the proposed one-time change to the surveillance frequency interval for RTBs.

4.0 EXIGENT CIRCUMSTANCES

Section 50.91(a)(6) states that where the NRC finds that exigent circumstances exist, in that a licensee and the NRC must act quickly and that time does not permit the publishing of a *Federal Register* Notice allowing 30 days for prior public comment, and it also determines that the amendment involves no significant hazards considerations, the NRC will either issue a *Federal Register* Notice providing for a limited period of opportunity for public comment or will utilize alternate means of communication as necessary to allow for public comment. The NRC will also require the licensee to explain the exigency and why the licensee cannot avoid it.

The following analysis was provided by the licensee in its letter dated January 20, 2009:

> Previous performances of SR 3.3.1.4 have been successful until this situation unexpectedly occurred on January 8, 2009. Upon discovery of this issue, Duke contacted the NRC to verbally provide all relevant available information. Catawba management took appropriate action in support of a resolution to this issue. This license amendment request was developed and submitted after this situation occurred. However, due to the impending expiration of the SR 3.3.1.4 frequency on February 19, 2009, insufficient time exists for processing this amendment request through normal channels. Sufficient time exists for processing this amendment through exigent channels as delineated in 10 CFR 50.91(a)(6).

Exigent circumstances exist in this situation because conduct of SR 3.3.1.4 under existing conditions may generate a reactor trip. Furthermore, if SR 3.3.1.4 is not conducted, then the licensee will have to begin the process of reactor shutdown to comply with their TSs. Therefore, the licensee proposes to conduct this SR after the appropriate repairs are made during their next planned outage. The NRC staff has reviewed the licensee's analysis and, based on this review, concluded that exigent circumstances do exist.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92(c), "Issuance of amendment," state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in its letter dated January 20, 2009:

> Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?
>
> Response: No.
>
> The Reactor Trip System (RTS) serves as accident mitigation equipment and is not required to function unless an accident occurs. The reactor trip bypass breakers are utilized to support testing of the reactor trip breakers (RTBs) while at power. This equipment does not affect any accident initiators or precursors. The proposed extension of the Technical Specification (TS) Surveillance Requirement (SR) 3.3.1.4 Frequency for RTBs does not affect its interaction with any system whose failure or malfunction could initiate an accident. Therefore, the probability of an accident previously evaluated is not significantly increased.
>
> The risk evaluation performed in support of this amendment request demonstrates that the consequences of an accident are not significantly increased. As such, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?
>
> Response: No.
>
> This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of the NRC granting of this proposed change. No changes are being made to the plant which will introduce any new or different accident causal mechanisms.
>
> Does the proposed amendment involve a significant reduction in the margin of safety?
>
> Response: No.
>
> Based on the availability of the RTS equipment and the low probability of an accident, Catawba concludes that the proposed extension of the surveillance test interval does not result in a significant reduction in the margin of safety.
>
> The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be significantly impacted by the proposed change. The risk implications of this request were evaluated and found to be acceptable.

The NRC staff has reviewed the licensee's analysis and, based on this review, has concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that the proposed amendment involves no significant hazards consideration.

6.0 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.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 changes 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 made a final finding that the amendment involves no significant hazards consideration. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated or, (b) create the possibility of a new or different kind of accident from any previously evaluated or, (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. Letter from J. R. Morris to the U.S. Nuclear Regulatory Commission, "Application for Amendment to Facility Operating License, One-Time Surveillance Requirement Frequency Extension for Reactor Trip Breakers," January 20, 2009 (ADAMS Accession number ML090220324)

2. 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 1," November 2002.
3. USNRC, Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998

4. 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

5. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007

6. NUREG-0800, Standard Review Plan 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, June 2007

7. USNRC, NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decision Making: Technical Specifications," Revision 1, March 2007

8. Letter, USNRC to J. R. Morris, "Catawba Nuclear Station, Units 1 and 2 – Issuance of Amendment Re: Reactor Trip System and Engineered Safety Features Actuation System Completion Times, Bypass Test Times and Surveillance Test Intervals (TAC Nos. MD 7718 and MD 7719) (ADAMS Accession number ML083460216)

Principal Contributors: J. Patel, NRR
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Date: February 13, 2009

February 13, 2009

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

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2, ISSUANCE OF AMENDMENT REGARDING ONE-TIME EXTENSION OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 3.3.1.4 FREQUENCY (TAC NO. ME0403)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 242 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Unit 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 20, 2009, as supplemented by letter dated February 5, 2009.

The amendment would allow a one-time limited duration extension of the TS Surveillance Requirement (SR) 3.3.1.4 frequency. SR 3.3.1.4 is a Trip Actuating Device Operational Test of the reactor trip breakers and reactor trip bypass breakers.

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

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

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

Docket No. 50-414

Enclosures:
1. Amendment No. 242 to NPF-52
2. Safety Evaluation

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ADAMS Accession No. ML090420426 *SE input provided ML090360378 **SE input provided by letter 2/11/09

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NAME	JThompson	MO'Brien	RElliott	AHowe*	WKemper**	LSubin	MWong	JStang
DATE	2/11/09	2/11/09	2/11/09	2/6/09	2/11/09	2/12/09	2/13/09	2/12/09

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