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Carolinas

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July 14, 2008

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC
Catawba Nuclear Station, Unit 1
Docket Number 50-413
Proposed Technical Specification Amendment
Technical Specification 3.6.6, Containment Spray System; 3.7.5, Auxiliary Feedwater System

Pursuant to 10 CFR 50.4, 10 CFR 50.90, and 10 CFR 50.91(a)(5), the licensee for Catawba Nuclear Station proposes a one-time limited duration extension of the Technical Specifications (TS) 3.6.6, Containment Spray System (CSS); and TS 3.7.5, Auxiliary Feedwater (AFW) System for Unit 1. These extensions are required to facilitate repair and replacement of the 1B NSWS pump and the activities associated with the repair.

On July 12, 2008 at approximately 1041 hours the 1B NSWS pump was declared inoperable and the 72 hour action statement of TS 3.7.8 was entered. On Sunday July 13, 2008, at approximately 1115 Operations realigned NSWS on Unit 1 utilizing operating procedure, OP/0/A/6400/006 C, Nuclear Service Water System, Enclosure 4.12B. This enclosure isolates NSWS flow to the 1B AFW pump, 1B CSS heat exchanger and the Unit 1 nonessential NSWS header. This allowed the "B" NSWS train to be declared operable and both units exited TS 3.7.8. This required Unit 1 to enter TS 3.6.6 Required Action A and TS 3.7.5 Required Action B, each with a 72 hour Completion Time. These TS Required Actions will remain in effect until the repairs and restoration of the 1B NSWS pump are complete. At that time, NSWS will be realigned to these components and the applicable Required Actions exited.

Although efforts are underway to replace the 1B NSWS pump will not be restored to operable status prior to expiration of the completion time. In order to avoid the shutdown of Catawba Unit 1, Duke proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1B AFW pump and the 1B CSS. The requested extension would allow continued operation of Unit 1 for an additional 144 hours while repairs and related testing of the 1B NSWS pump are completed.

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Both units are currently at 100% power. Completion Times of the applicable Required Actions expire on July 15, 2008 at 1041 hours. An estimated repair time for the 1B NSWS pump is nine (9) days and thus this will exceed the 72 hours allowed by the TS. Therefore, in order to avoid the shutdown of Catawba Unit 1, Duke requests approval of this license amendment on a one-time emergency basis by July 15, 2008 at 0800 hours.

Attachment 1 provides a description of the proposed change and technical justification, an evaluation of significant hazards consideration pursuit to 10 CFR 50.92 (c) and an environmental assessment.

Attachment 2 provides the existing TS pages marked-up to show the proposed change.

Attachment 3 contains retyped (clean) TS pages.

Attachment 4 lists the regulatory commitments documented in this request.

Attachment 5 contains a Catawba PRA quality discussion.

In accordance with Duke Energy Corporation administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the Catawba Plant Operations Review Committee and the Corporate Nuclear Safety Review Board.

Implementation of this amendment request will not require changes to the Catawba Updated Final Safety Analysis Report (UFSAR).

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate State of South Carolina official.

Should you have any questions concerning this information, please call A. P. Jackson at (803) 701-3742.

Very truly yours,

James R. Morris

Attachments

James R. Morris affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

James R. Morris,
Site Vice President

Subscribed and sworn to me: July 14, 2008
Date

Notary Public

My commission expires: 1/31/2018
Date

SEAL

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NCEMC
PMPA
SREC
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ATTACHMENT 1

DESCRIPTION OF PROPOSED CHANGES, TECHNICAL JUSTIFICATION, SIGNIFICANT HAZARDS CONSIDERATION PURSUANT TO 10CFR50.92(C) AND ENVIRONMENTAL ASSESSMENT

1. Description:

Pursuant to 10 CFR 50.90 and 10 CFR 50.91 (a) (5), Duke Energy Carolinas, LLC (Duke), the licensee for Catawba Nuclear Station, proposes a one-time limited duration extension of the Technical Specification (TS) 3.7.5 Required Action B.1 Completion Time associated with the 1B Auxiliary Feedwater (AFW) pump, and TS 3.6.6 Required Action A.1 Completion Time associated with the 1B Containment Spray System (CSS). The requested extension would allow continued operation of Unit 1 for an additional 144 hours above the 72 hour action statement time while repairs and related testing of the 1B nuclear service water system (NSWS) pump are completed.

The proposed amendment is being requested on an emergency basis pursuant to 10 CFR 50.91 (a) (5). On July 12, 2008 at approximately 1041 hours the 1B NSWS pump was declared inoperable and the 72 hour action statement of TS 3.7.8 was entered. On Sunday July 13, 2008, at approximately 1115 Operations realigned NSWS on Unit 1 utilizing operating procedure, OP/0/A/6400/006 C, Nuclear Service Water System, Enclosure 4.12B. This enclosure isolates NSWS flow to the 1B AFW pump, 1B CSS heat exchanger and the Unit 1 nonessential NSWS header. This allowed the "B" NSWS train to be declared operable and both units exited TS 3.7.8. This required Unit 1 to enter TS 3.6.6 Required Action A and TS 3.7.5 Required Action B each with a 72 hour Completion Time. These TS Required Actions will remain in effect until the repairs and restoration of the 1B NSWS pump are complete. At that time, NSWS will be realigned to these components and the applicable Required Actions exited.

Efforts are currently in progress to replace the 1B NSWS pump; however, the repairs will not be completed prior to expiration of the current completion time at 1041 on July 15, 2008. Therefore, in order to avoid the shutdown of Catawba Unit 1, Duke requests approval of this license amendment application on a one-time emergency basis by July 15, 2008 at 0800.

2. Proposed Change:

The proposed change would add two new License Conditions to Appendix B of the Catawba Nuclear Station Unit 1 Facility Operating License, License Number NPF-35. The proposed License Conditions are as follows

- The 72 hour allowed outage time of Technical Specification 3.7.5 Action "B" for the 1B AFW pump which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008.

- The 72 hour allowed outage time of Technical Specification 3.6.6 Action "A" for the 1B CSS which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008.

3. Background:

The NSWS, including Lake Wylie and the Standby Nuclear Service Water Pond (SNSWP), provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the NSWS also provides this function for various safety related and non-safety related components.

The NSWS consists of two independent loops (A and B) of essential equipment, each of which is shared between units. Each loop contains two NSWS pumps, each of which is supplied from a separate emergency diesel generator. Each set of two pumps supplies two trains (1A and 2A, or 1B and 2B) of essential equipment through common discharge piping. While the pumps are unit designated, i.e., 1A, 1B, 2A, 2B, all pumps receive automatic start signals from a safety injection or blackout signal from either unit. Therefore, a pump designated to one unit will supply post accident cooling to equipment in that loop on both units, provided its associated emergency diesel generator is available. For example, the 1A NSWS pump, powered by emergency diesel 1A, will supply post accident cooling to NSWS trains 1A and 2A.

An NSWS train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. 1. Both NSWS pumps on the NSWS loop are OPERABLE;

-or-

2. One unit's NSWS pump is OPERABLE and one unit's flow path to the non essential header, AFW pumps, and Containment Spray heat exchangers are isolated (or equivalent flow restrictions);

-and-

b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE:

If a shared NSWS component becomes inoperable, or normal or emergency power to shared components becomes inoperable, then the Required Actions of this LCO must be entered independently for each unit that is in the MODE of applicability of the LCO, except as noted in a.2 above.

The NSWS has another safety related function with regard to the AFW system. The condensate storage system supplies the AFW system suction source requirements during normal system operating modes; but, since the condensate storage system is not safety related its availability is not assured. The assured source of water supply to the AFW pumps is provided by the safety related portion of the Nuclear Service Water System.

Another safety related function of the NSWS is to supply cooling water to the CSS heat exchangers during the recirculation phase of a loss of coolant accident. In the recirculation mode of operation, containment spray pump suction is transferred from the refueling water storage tank (RWST) to the containment recirculation sump(s). When the containment spray system suction is from the containment recirculation sump, its associated heat exchanger receives NSWS flow for cooling.

On July 12, 2008 at approximately 0910 the control room operators started the 1B NSWS pump and stopped the 1A NSWS pump in support of a NSWS train B supply header flush. At approximately 1041 the control room operators received several NSWS alarms for B NSWS train header pressure and flow. In addition, the operators observed low discharge header pressure along with high NSWS flow for the 2B NSWS pump and low flow for the 1B NSWS pump. The operators entered their abnormal procedure for the NSWS and started the 1A NSWS pump and stopped the 1B NSWS pump. The 1B NSWS pump was declared inoperable as of 1041 and both units entered TS 3.7.8 Action A with a 72 hour completion time.

On Sunday July 13, 2008, at approximately 1115 Operations realigned NSWS on Unit 1 utilizing operating procedure, OP/0/A/6400/006 C, Nuclear Service Water System, Enclosure 4.12B. This enclosure isolates NSWS flow to the 1B AFW pump, 1B CSS heat exchanger and the Unit 1 nonessential header. This allowed the "B" NSWS train to be declared operable and both units exited TS 3.7.8. This required Unit 1 to enter TS 3.6.6 Required Action A and TS 3.7.5 Required Action B each with a 72 hour Completion Time. These alignments allow the 2B NSWS pump to carry the loads for Unit 1 Train B except for the isolated sections discussed above. Since the required flows are not available for the 1B AFW pump and the 1B CSS heat exchangers, the start time of the LCO reverts back to the time the NSWS was taken out of service originally. These TS Required Actions will remain in effect until the repairs and restoration of the 1B NSWS pump are complete. At that time, NSWS will be realigned to these components and the applicable Required Actions exited.

Completion times for the applicable TS Required Actions for the 1B AFW pump and 1B CSS train expire at 1041 on July 15, 2008. Although efforts are underway to replace the 1B NSWS pump, the pump will not be restored to operable status prior to expiration of the completion time.

In order to avoid the shutdown of Catawba Unit 1, Duke proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1B AFW pump and the 1B CSS. The requested extension would allow continued operation of Unit 1 for an additional 144 hours while repairs and related testing of the 1B NSWS pump are completed.

4. <u>Current Requirements</u>:

TS 3.7.5, "Auxiliary Feedwater (AFW) System" contains LCO 3.7.5. This LCO governs the AFW system for Modes 1, 2, 3, and 4 when steam generator is relied upon for heat removal. LCO 3.7.5 requires three AFW trains to be operable. Condition B for this LCO states that with one AFW train inoperable, the inoperable train must be restored to operable status within 72 hours. Condition C states that if the required action and associated completion time is not met, the unit must be in Mode 3 within 6 hours and in Mode 4 within 12 hours.

TS 3.6.6, "Containment Spray System" contains LCO 3.6.6. This LCO governs the CSS for Modes 1, 2, 3, and 4. LCO 3.6.6 requires two CSS trains to be operable. Condition A for this LCO states that with one CSS train inoperable, the inoperable train must be restored to operable status within 72 hours. Condition B states that if the required action and associated completion time is not met, the unit must be in Mode 3 within 6 hours and in Mode 5 within 36 hours.

5. <u>**Basis for Current Requirements:**</u>

<u>LCO 3.7.5 Basis Discussion</u>

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%. In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation valve leakage and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB);
b. Loss of Main Feedwater (MFW)

In addition, the minimum available AFW flow and system characteristics are considered in the analysis of a small break loss of coolant accident (LOCA) and events that could lead to steam generator tube bundle uncovery for dose considerations.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36.

<u>LCO 3.6.6 Basis Discussion</u>

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment engineered safety feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the Air Return System (ARS) being rendered inoperable.

The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to be within the containment environmental qualification temperature during the DBA SLB. The basis of the containment environmental qualification temperature is to ensure the OPERABILITY of safety related equipment inside containment.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36.

6. Reason for Requesting Emergency Amendment:

Regulation 10 CFR 50.91(a) (5) states that where the NRC finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention or either resumption of operation or increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. The regulation also states that the NRC will decline to dispense with notice and comment on the no significant hazards if it determines that the licensee has abused the emergency provision by failing to make timely application for the amendment and thus itself creating the emergency. The regulation requires that a licensee requesting an emergency amendment explain why the emergency situation occurred and why the licensee could not avoid the situation. As explained below, an emergency amendment is needed to preclude a plant shutdown and cooldown, and Duke could not have reasonably avoided the situation or made timely application for an amendment.

7. Reason the Emergency Situation Has Occurred:

On July 12, 2008 at approximately 0910 the control room operators started the 1B NSWS pump and stopped the 1A NSWS pump in support of NSWS train B supply header flushes. At approximately 1041 the control room operators received several NSWS alarms for B NSWS train header pressure and flow. The operators observed low discharge header pressure along with high NSWS flow for the 2B NSWS pump and low flow for the 1B NSWS pump. The operators entered their abnormal procedure for the NSWS and started the 1A NSWS pump and stopped the 1B NSWS pump. The 1B NSWS pump was declared inoperable as of 1041 and both units entered TS 3.7.8 Action A with a 72 hour completion time.

On Sunday July 13, 2008, at approximately 1115 Operations realigned NSWS on Unit 1 utilizing operating procedure, OP/0/A/6400/006 C, Nuclear Service Water System, Enclosure 4.12B. This enclosure isolates NSWS flow to the 1B AFW pump, 1B CSS heat exchanger and the Unit 1 nonessential header. This allowed the "B" NSWS train to be declared operable and both units exited TS 3.7.8. This required Unit 1 to enter TS 3.6.6 Required Action A and TS 3.7.5 Required Action B each with a 72 hour Completion Time. These TS Required Actions will remain in effect until the repairs and restoration of the 1B NSWS pump are complete. At that time, NSWS will be realigned to these components and the applicable Required Actions exited.

Completion times for the applicable TS Required Actions for the 1B AFW pump and 1B CSS train expire at 1041 on July 15, 2008. Although efforts are underway to repair the 1B NSWS pump and it will not be restored to operable status prior to expiration of the completion time.

In order to avoid the shutdown of Catawba Unit 1, Duke proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1B AFW pump and the 1B CSS. The requested extension would allow continued operation of Unit 1 for an additional 144 hours while repairs and related testing of the 1B NSWS pump are completed.

Both units are currently at 100% power. An estimated repair time for the 1B NSWP is nine (9) days and thus this will exceed the 72 hours allowed by the TS. Therefore, in order to avoid the shutdown of Catawba Unit 1, Duke requests approval of this license amendment on a one-time emergency basis by July 15, 2008 at 0800 hours.

8. Reason the Situation Could Not Have Been Avoided:

Initial Incident Investigation

At 0030, on July 13, 2008, a diver crew and maintenance pump team performed an inspection of the pump. The diver crew entered the suction pit and discovered several metallic pieces lying on the bottom of the pump house pit floor. The diver crew retrieved the pieces for further inspection. While on location at the entrance to the suction bell, the maintenance pump team proceeded to hand rotate the pump. The diver crew did not identify any movement in the first stage impeller while the pump crew successfully rotated the shaft at the pump and motor coupling. Therefore, it was evident the impeller assembly was no longer connected to the motor shaft.

The Nuclear Service Water (NSWS) Pump is a deep draft vertical pump. It is a 1000 HP, two stage Bingham-Willamette VTM 30 x 44C pump. It is assembled with a suction bell, two bowl assemblies, four columns, one discharge head and motor to make the complete vertical assembly approximately 65 feet tall. It consists of five shafts and correspondingly four couplings. Only the uppermost motor to pump head shaft is accessible without complete pump removal and disassembly. Therefore complete removal is necessary for further investigation and repair.

NSWS Pump Monitoring

The Catawba NSWS pumps are high safety significance pumps and receive in depth monitoring, trending, and analysis.

1) Vibration: Data collected quarterly via IWP and Maintenance testing programs. Amplitude, frequency, and time waveforms are reviewed in detail for any changes. Data is reviewed by Category III and IV certified vibration analysts. The most recent vibration data collected on the 1B NSWS pump was collected just prior to the recent 1EOC 17 RFO. No abnormal data was evident.

2) Pump Pressure and Flow: Suction and discharge pressure as well as flow is monitored closely on a quarterly basis via required procedural IWP testing. Suction and discharge pressure have remained within acceptable values.

3) Oil Analysis Data: Motor oil samples are collected on a quarterly basis. No significant changes in oil quality have been noted.

4) Preventive Maintenance: Pump assemblies are replaced on a periodic frequency based on vendor recommendations and industry experience for this type of pump. The pump was last refurbished is 2003. Per the preventative maintenance program the next overall/rebuild should not be needed until 2015.

A comprehensive review of the previous 4 quarters of in-service test data for the 1B NSWS Pump was reviewed to verify that no performance degradation had occurred prior to the failure of the pump on July 12, 2008. The pump flow rate and discharge pressure were well within the established acceptance criteria. The pump motor inboard and outboard vibration readings were well below the acceptance criteria. There were no negative trends noted on any of the measured parameters. A review of previous work history on the 1B NSWS Pump did not identify any work activity generated as a result of degrading pump conditions indicative of impending coupling failure. Therefore, the failure of the pump coupling was not predictable based on the quarterly test data. When completed, the results of the Root Cause Investigation will be incorporated into the NSWS pump monitoring program.

Additional Actions

Based on the above discussion, Catawba has been actively monitoring pump data and this failure could not have been predicted. Neither a routine nor an exigent TS amendment request could have been processed within the 72 hour period. Therefore, an emergency TS amendment is required to preclude a shutdown.

9. **Technical Evaluation:**

Extent of Condition Discussion

This is a proven pump design with an excellent operating history. NSWS Pumps are being changed out due to aging and have not had a history of failures or operational issues. This pump was completely refurbished in 2003. Each of the NSWS pumps have been refurbished twice during the operation of Catawba with only normal wear identified during the refurbishments/inspections. There have been no design changes to the pumps that would create a common mode failure. The A NSWS pump pit was verified free of foreign material in May 2008 during the unit 1 refueling outage. The B NSWS pump pit was inspected on July 12, 2008. The only foreign material identified in the B pit was associated with the failed coupling. Operating and maintenance practices have not changed. Therefore, it is concluded that this failure is not transportable to the other pumps.

The NSWS pump changeout schedule is shown below:

NSWS Pump	Date of Last Changeout	Date of Next Scheduled Changeout
1A	2008	2020
1B	2003	Failed, Root Cause
2A	2004	2018
2B	1998	2012
Spare	1991	2009 (1B replacement)

* Duke plans to changeout the 1B pump at the next refueling outage, which is the 1EOC18 outage in November 2009.

Condition of 1A NSWS Pump

The 1A NSWS pump was refurbished with a new rotating element during the Unit 1 refueling outage in May/June 2008. The pump was subsequently tested following replacement during this refueling outage and verified to meet its flow requirements for single pump and dual pump alignments. The performance parameters of the 1A NSWS pump indicate the pump is in good running condition and considered reliable for many years of service. The next planned overhaul is the refueling outage in 2020. In addition to the overhaul completed during the refueling outage in May/June, 2008, the "A" NSWS pit was inspected for cleanliness of the suction intake of both the 1A and 2A Nuclear Service Water Pumps. During this 1B pump replacement, the 1A NSWS pump and its support systems will be considered protected equipment. No scheduled maintenance will be performed on those systems.

Condition of 2A NSWS Pump

The 2A NSWS pump was refurbished with a new rotating element during the Unit 2 refueling outage in November 2004. The pump was subsequently tested following replacement during this refueling outage and verified to meet its flow requirements for single pump and dual pump alignments. Subsequently, since the November 2004 overhaul, the pump performance parameters are verified on a quarterly basis per the In-service Testing Requirements. The results indicate the 2A NSWS pump is in good running condition and considered reliable for many years of service. The next planned overhaul is the refueling outage in 2018. During the refueling outage in May, 2008, the "A" NSWS pit was inspected for cleanliness of the suction intake of both the 1A and 2A Nuclear Service Water Pumps. During this 1B pump replacement, the 1A NSWS pump and its support systems will be considered protected equipment. No scheduled maintenance will be performed on those systems.

Condition of the Spare Pump for 1B

The replacement pump for the failed 1B Nuclear Service Water Pump is the 1A Pump which was removed from service during the refueling outage (1EOC17) in May/June, 2008. The 1A Pump was operating well and removed from service due to age as part of the station's pump overhaul/replacement plan for equipment reliability. The 1A Pump performance prior to removal from service has been specifically reviewed to determine acceptable service for installation as the replacement 1B pump. The In-service Test data from the previous four tests, specifically the flow rate, discharge pressure and pump vibration data demonstrate the pump will operate smoothly within the test acceptance criteria. Prior to installation of the spare pump, an inspection of the assembly will be performed to assess parts requiring refurbishment/replacement. This pump will be replaced in next refueling outage in November 2009 (1EOC18). While this pump is being replaced, the NSWS pit B will be drained and a cleanliness inspection will be performed. Once in service, the 1B NSWS pump will be operated to the extent practicable. This pump will be in the normal equipment rotation and will be operated as required to support routine train maintenance activities.

Condition of the 2B NSWS Pump

The 2B NSWS pump was refurbished with a new rotating element during the Unit 2 refueling outage in September 1998. The pump was subsequently tested following replacement during this refueling outage and verified to meet its flow requirements for single pump and dual pump alignments. Subsequently, since the September 1998 overhaul, the pump performance parameters are verified on a quarterly basis per the In-service Testing Requirements. The results indicate the 2B NSWS pump is in good running condition and considered reliable for many years of service. The next planned overhaul is the refueling outage in 2012. While the 1B pump is being replaced, the NSWS pit B will be drained and a cleanliness inspection will be performed for the suction intake of both the 1B and 2B Nuclear Service Water Pumps.

Testing Requirements:

Following the replacement of the 1B NSWS pump, the pump will be tested for operational readiness in accordance to the 1998 ASME Code. The 1998 ASME Code for the In-service Test Requirements mandate the pump to be tested for performance flow and pressure parameters and axial and radial vibration. These parameters are tested quarterly per the program requirements. The results are evaluated against Acceptable, Alert or Unacceptable Limits. These tests include head curve verification, flow, pressure, vibration followed by train related flow balance which confirms operability.

Current Plant Status

At the time of the incident NSWS pipe work on buried piping was in progress and various locations were uncovered for pipe inspections. Currently, this work has been put on hold and the buried piping has been covered per the requirements for tornado missile protection.

Additional Discussion:

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design basis spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Feature (ESF) bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The RWST is protected by a missile proof barrier wall which ensures a sufficient quantity of refueling water is retained in the tank to allow for an emergency cooldown in the event the tank is punctured by a missile. The RWST is a safety related seismic category I structure.

When the containment spray system suction is from the containment recirculation sump, its associated heat exchanger receives NSWS flow for cooling. During the extended time period this flow will not be available. However this does not affect the initial injection flow provided.

There are several sources of water available to the AFW pumps. The preferred sources are non-safety grade condensate quality, located in the Turbine and Service Buildings. These are called the condensate storage system. The condensate storage system is formed from the Upper Surge Tanks (two 42,500 gallon tanks per unit) and the Condenser Hotwell (normal operating level of 170,000 gallons). The condensate storage system supplies the AFW requirements during normal system operating modes; but, since the condensate storage system is not safety related its availability is not assured. The assured source of supply to the AFW pumps is provided by the safety related portion of the Nuclear Service Water System.

TS 3.7.6 requires the condensate storage system to be operable in modes 1, 2, 3 and mode 4 when steam generators are relied upon for heat removal. The condensate storage system contains sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100% Rated Thermal Power (RTP), and then to cool down the reactor coolant system (RCS) to RHR entry conditions, assuming a natural circulation cooldown. In doing this, it retains sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

Another non-safety grade source of condensate water for the AFW pumps is the Auxiliary Feedwater Condensate Storage Tank (CACST). Each unit has a CACST that is maintained full by a recirculation flow of condensate from the condensate system and overflow to the condensate storage system. The CACST holds approximately 42,500 gallons of condensate grade water.

For emergency events, when none of the condensate grade sources are available, two redundant and separate trains of nuclear service water are available. The water supplied by the two nuclear service water sources is of lower quality; however, safety considerations override those of steam generator cleanliness. The NSWS assured source of water supply is configured into two trains. The turbine driven AFW pump receives NSWS from both trains of NSWS, therefore, the loss of one train of assured source renders only one AFW train inoperable. The remaining NSWS train provides an operable assured source to the other motor driven pump and the turbine driven pump. Therefore, during the extended time period, the 1B AFW pump will be capable of starting and providing water to the steam generators from the non-safety sources and only its safety-related source from the NSWS will be affected.

Therefore, Catawba is requesting an extension of the Completion time to support repair of the 1B NSWS pump. The plant configuration during this time frame will still be able to support Chapter 15 accident analysis. The probabilistic risk assessment discussed below describes the effect of the extension of the Completion time.

Risk Evaluation

Duke has used a risk-informed approach to determine the risk significance of extending the current Technical Specification associated with the 1B Nuclear Service Water System (NSWS) pump work. The Unit 1 extension is for an additional 144 hours for a total time of 216 hours.

The cumulative risk impact for this evolution is the sum of Part 1 and Part 2 risk numbers. This is shown below:

Catawba Unit 1

Unit 1	Part 1 (3 days)* (2 RN pumps)	Part 2 (3 days)** (1 RN pump)	Total (Part 1+Part 2) 6 days
dCDF/yr	6.9E-07	4.5E-07	**1.1E-06**
dLERF/yr	2.7E-08	1.4E-08	4.1E-08
	Part 1 (3 days)* (2 RN pumps)	Part 2 (6 days)*** (1 RN pump)	Total (Part 1+Part 2) 9 days
ICCDP	6.9E-07	8.9E-07	**1.6E-06**
ICLERP	2.7E-08	2.8E-08	5.5E-08

*3 days beyond the original 72 hr TS CT
**3 days of the 6 days extension period beyond the original 72 hr TS CT
***6 days representing the original 72 hr TS CT plus 3 days beyond the original 72 hr TS CT

The delta CDF associated with the 6 day extension related to the NS and CA assured source is approximately 1.1E-06. The result is slightly above the RG-1.174 guidance of 1.0E-06. The ICCDP is estimated to be 1.6E-06. The result is above the RG-1.177 guidance (5E-07) for a permanent TS change.

The LERF results are less limiting than the CDF results.

Dominant Sequences

The dominant sequences are reactor coolant pump seal LOCAs that occur when all RN is lost as an initiating event (dominated by some common cause failure of the available RN pumps); failure to restore cooling to the RCP seals (both SSF and YD) with failure to trip the RCPs prior to seal failure.

The dominant SSF failure is a failure to activate the SSF in time (human error).

The dominant YD failures are the human error of failure to activate or the inability to align YD because YD has been aligned to the other unit.

Impact of PRA Analysis on Fire and Flooding Events
There were few fire initiated cut sets above the CDF and LERF truncation limits. Additionally there were few flood cut sets. Fires and floods contributed negligibly to the CDF and LERF results.

RG 1.200 Assessment
In accordance with the ASME standard [Reference 6] and RG 1.200 [Reference 5] Duke has made an assessment of all the ASME Supporting Requirements (SRs).

The Catawba PRA fully meets 224 of the 306 ASME PRA Standard Supporting Requirements (SRs), as modified by Reg. Guide 1.200. In addition, 24 of the SRs are not applicable to the Catawba PRA, either because the referenced techniques are not utilized in the PRA or because the SR is not required for Capability Category II.

Of the 58 open SRs, 14 are of a technical nature. The remaining open SRs require enhanced documentation. However, none of the open items are expected to have a significant impact on the PRA results or insights, as discussed in Attachment 5 of this document.

PRA Model
The Catawba PRA is a full scope PRA including both internal and external events. The model includes the necessary initiating events (e.g., LOCAs, transients) to evaluate the frequency of accidents. The previous reviews of the Catawba PRA, NRC and peer reviews, have not identified deficiencies related to the scope of initiating events considered.

The Catawba PRA includes models for those systems needed to estimate core damage frequency. These include all of the major support systems (e.g., ac power, service water, component cooling, and instrument air) as well as the mitigating systems (e.g., emergency core cooling). These systems are generally modeled down to the component level, pumps, valves, and heat exchangers. This level of detail is sufficient for this application.

Truncation Limit
Truncation issues are not an issue with this risk calculation. The analysis for the current configuration was performed at the same truncation level as the base case (5.0E-10 for CDF and 5.0E-11 for LERF). A review of the cut sets shows that loss of nuclear service water with a failure of drinking water backup cooling to the "A" charging pump with a corresponding failure to initiate the SSF are in most of the top cut sets. There is adequate representation of the expected failure in the results that drive the answer so that there was no need to solve to any lower truncation levels. The issue identified in RG 1.177 (most of the failures appearing near the truncation cutoff) does not exist in this analysis. Additionally, an explicit

truncation level analysis was performed for Revision 3a of the PRA consistent with ASME standard and RG 1.200 requirements.

Uncertainty and Sensitivity

Duke agrees with the RG 1.177 statement that risk analyses of CT extensions are relatively insensitive to uncertainties. The PRA did not credit equipment repair so there are no uncertainties to be evaluated for that issue. Important systems are required to remain in service during the CT so no issues with mean downtimes should exist. Thus uncertainty and sensitivity are not expected to alter the conclusions of the evaluation.

Results of Reviews with Respect to this LAR

A review of the analyses (cut sets and pertinent accident sequences) was made for accuracy and completeness. Specifically, cut sets generated for the solutions were screened and invalid cut sets were removed and appropriate recovery events applied. This process is documented in Duke calculations. The review verified that the calculations adequately modeled the effects of the NSWS system unavailability.

Consistent with the work place procedures governing PRA analysis, this calculation has undergone independent checking by a qualified reviewer. Additionally the Catawba Plant Operations Review Committee (PORC) and Duke Nuclear Safety Review Board (NSRB) reviewed and approved this amendment request package.

Tier 2 Assessment: Avoidance of Risk-significant Plant Equipment Outage Configurations

Tier 2 provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change. Specific components and trains have been identified that are not to be taken out of service during the period of the extended CT.

Duke has several Work Process Manual procedures and Nuclear System Directives that are in place at Catawba Nuclear Station to ensure that risk-significant plant configurations are avoided. The key documents are as follows:

- Nuclear System Directive 415, "Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a.4)".
- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)".
- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL".
- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL".

The proposed changes are not expected to result in any significant changes to the current configuration risk management program. The existing program uses a blended approach of quantitative and qualitative evaluation of each configuration assessed. The Catawba on-line computerized risk tool, ORAM-Sentinel, considers both internal and external initiating events with the exception of seismic events. Thus, the overall change in plant risk during maintenance activities is expected to be addressed adequately in accordance with RG 1.177 considering the proposed Technical Specifications.

Tier 3 Assessment: Maintenance Rule Configuration Control
10 CFR 50.65(a)(4), RG 1.182, and NUMARC 93-01 require that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increase in risk that may result from proposed maintenance activities. These requirements are applicable for all plant modes. NUMARC 91-06 requires utilities to assess and manage the risks that occur during the performance of outages.

As stated above, Duke has approved procedures and directives in place at Catawba to ensure the requirements of the Maintenance Rule are implemented. These documents are used to address the Maintenance Rule requirements, including the on-line (and off-line) Maintenance Policy requirement to control the safety impact of combinations of equipment removed from service.

More specifically, the Nuclear System Directives address the process, define the program, and state individual group responsibilities to ensure compliance with the Maintenance Rule. The Work Process Manual procedures provide a consistent process for utilizing the computerized software assessment tool, ORAM-SENTINEL, which manages the risk associated with equipment inoperability.

ORAM-SENTINEL is a Windows-based computer program designed by the Electric Power Research Institute as a tool for plant personnel to use to analyze and manage the risk associated with all risk significant work activities including assessment of combinations of equipment removed from service. It is independent of the requirements of Technical Specifications and Selected Licensee Commitments.

The ORAM-SENTINEL models for Catawba are based on a "blended" approach of probabilistic and traditional deterministic approaches. The results of the risk assessment include a prioritized listing of equipment to return to service, a prioritized listing of equipment to remain in service, and potential contingency considerations.

Additionally, prior to the release of work for execution, Operations personnel must consider the effects of severe weather and grid instabilities on plant operations. This qualitative evaluation is inherent of the duties of the Work Control Center Senior Reactor Operator (SRO). Responses to actual plant risk due to severe weather or grid instabilities are programmatically incorporated into applicable plant emergency or response procedures.

Previous NRC RAIs on PRA Model

Duke reviewed previous requests for additional information from a previous emergency TS submittal and provides the following responses:

Question 1:
The submittal identified administrative controls to assure plant changes are reflected in the PRA model, but has not stated whether there are outstanding plant changes not yet reflected in the model, and whether those would impact this analysis.

Response:
All outstanding plant changes that are not included in the current base PRA model (Rev. 3a) were reviewed and evaluated for this application. Of this population, 6 plant changes were determined to require further evaluation. These are summarized below:

Issue	Resolution
Closure of two valves important to ISLOCA sequences may not occur.	ISLOCA is important for LERF sequences. LERF is not the limiting metric. Additionally ISLOCA sequences for LERF were not the dominant sequences.
Add alternate feedwater makeup line to each S/G.	This would be a risk reduction. No action was taken so the model is conservative.
The chemical and volume control system model does not capture all of the unavailability for drinking water backup cooling to the "A: charging pump.	The model was revised to include new failure mechanism to reflect a 50-50 chance that an "A" charging pump will receive backup flow from drinking water.
Increase the exposure time for 6 basic events to reflect current testing.	Exposure time was increased for the 6 basic events based on current testing schedule.
Replace recovery in model with explicit logic.	Recovery was set equal to 1.0 in the model. This recovery event did not appear in dominant cut sets.
Modifications to the NSWS headers to add crossover between EDGs	Expected CDF improvements. Current model is bounding.

Question 2:
The submittal did not address truncation levels per RG 1.177 2.3.3.4.

Response:
Truncation issues are not an issue with this risk calculation. The analysis for the current configuration was performed at the same truncation level as the base case (5.0E-10 for CDF and 5.0E-11 for LERF). A review of the cut sets shows that loss of nuclear service water with a failure of YD backup cooling to the "A" NV pump with a corresponding failure to initiate the SSF are in most of the top cut sets. There is adequate representation of the expected failure in the results that drive the answer so that there was no need to solve to any lower truncation levels. The issue identified in RG 1.177 (most of the failures appearing near the truncation cutoff) does not exist in this analysis. Additionally, an explicit truncation level analysis was performed for Revision 3a of the PRA consistent with ASME standard and RG 1.200 requirements.

Question 3:
The submittal needs to identify if credit is taken for the SSF in the risk calculations, and should also address if equipment repair is credited.

Response:
Credit is taken for the SSF. Catawba has taken action to ensure that the SSF will be available during the extended CT period. The Catawba PRA does not take credit for equipment repair.

Question 4:
The submittal did not address uncertainty or sensitivity issues per RG 1.177 2.3.5.

Response:
Duke agrees with the RG 1.177 statement that risk analyses of CT extensions are relatively insensitive to uncertainties. We did not credit equipment repair so there are no uncertainties to be evaluated for that issue. We required important systems to remain in service during the CT so no issues with mean downtimes should exist. Therefore, for the typical issues related to uncertainties, there should be no effect on our analysis.

Question 5:
Provide clarification that the seismic contribution is negligible compared to the non-seismic results.

Response:
We have numerically reviewed the seismic impact for the nuclear service water system, including a loss of emergency diesel generator using the previous PRA model and determined that the seismic contribution is negligible compared to the

non-seismic results. Based on the expected configuration during the time period of the CT extension, there is no reason to expect that that conclusion would change for the current model.

References

1. E-Mail: Tony Jackson to Randy Hart, Subject: Potential Emergency TS Change for 1B RN Pump, July 13, 2008.
2. Phone Call: M.S. Kitlan, Jr, to Randy Hart, July 13, 2008.
3. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 1, November 2002.
4. U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Regulatory Guide 1.177, Revision 0, August 1998.
5. U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 1, January 2007.
6. American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME-RA-Sc-2007.
7. SAAG 592, Rev.2: Determination of Risk Significance when Taking an RN Loop Out of Service Beyond Tech. Spec. Limits, (Catawba PRA Rev 2a) August 2000.
8. SAAG 725, Risk Significance Determination for RN Piping Replacement Project (CNCE 71424) (Catawba PRA Rev 2b), August 2002.
9. SAAG 278, Catawba Rev. 3 Seismic PRA Analysis, November 2003.
10. CNC-1535.00-00-0025, Risk Significance Determination for Proposed Catawba RN Loop LCOs, Revision 3, April 2006.
11. Catawba PRA Revision 3a.
12. NUREG/CR-5497, Common Cause Failure Parameter Estimations, October 1998.
13. SAAG 710, Catawba RN Essential Header NOED, 2002.
14. SAAG 670, Catawba PRA Rev. 3 Common Cause Failure Analysis, Table 1, Revision 3, October 2005.
15. DPC-1535.00-00-0013, PRA Quality Self-Assessment, DRAFT.
16. TSAIL printout from site (see electronic documents).

Operation and Maintenance Restrictions for the Duration of the Extension

These items are listed in Attachment 4 to this document.

10. Regulatory Safety Analysis:

10.1 No Significant Hazards Consideration:

Duke has concluded that operation of the Catawba Nuclear Station Unit 1 in accordance with the proposed change to the Technical Specifications (TS) does not involve a significant hazards consideration. Duke's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a) (1), of the three standards set forth in 10 CFR 50.92 (c).

i. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The 1B AFW pump and the 1B CSS safety related functions are as accident mitigators and are not required unless an accident occurs. The 1B AFW pump and 1B CSS do not affect any accident initiators or precursors. The proposed extension of the Required Action Completion Time does not affect the 1B AFW pump's and 1B CSS 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 (Reference Section 9) 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.

ii. 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.

iii. Does the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

Based on the availability of redundant systems, the restrictions on maintenance and operation of required systems, and the low probability of an accident, Catawba concludes that the reduction of availability of the 1B AFW pump and the 1B CSS 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.

10.2 Applicable Regulatory Requirements/Criteria:

The analysis presented in this LAR demonstrates that Catawba will remain in compliance with the applicable regulations and requirements. These are:

10 CFR 50.46 and 10 CFR 50, Appendix A, General Design Criterion (GDC) 44,45 and 46.

This LAR is being submitted in accordance with 10 CFR 50.90 and 50.91 (a) (5).

11. Environmental Consideration:

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed changes meet the eligibility criteria for the categorical exclusion set forth in 10 CFR 51.22 (c) (9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed change is not required.

12. Precedent:

None

ATTACHMENT 2

MARKED-UP CATAWBA
TECHNICAL SPECIFICATION

Amendment Number	Additional Condition		Implementation Date
173	The schedule for the performance of new and revised surveillance requirements shall be as follows: For surveillance requirements (SRs) that are new in Amendment No. 173 the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment No. 173. For SRs that existing prior to Amendment No. 173, including SRs with modified acceptance criteria and SRs who intervals of performance are being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of amendment No. 173. For SRs that existed prior to Amendment No. 173, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of Amendment No. 173		By January 31, 1999
180	The maximum rod average burnup for any rod shall be limited to 60 GWd/mtU until the completion of an NRC environmental assessment supporting an increased limit.		Within 30 days of date of amendment.
	In association with the ECCS sump strainer modification and Generic Safety Issue (GSI)-191 requirements: 1. Unit 1 shall enter Mode 5 for the outage to install the sump strainer modification no later than May 19, 2008 and 2. The Unit 1 sump strainer modification shall be completed prior to entry into Mode 4 after May 19, 2008.		Within 30 days of date of amendment and no later than December 31, 2007

INSERT A

Renewed License No. NPF-

Amendment No. 237

Insert A

	• The 72 hour allowed outage time of Technical Specification 3.7.5 Action "B" for the 1B AFW pump which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008. • The 72 hour allowed outage time of Technical Specification 3.6.6 Action "A" for the 1B CSS which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008.		July 15, 2008 at 1041

ATTACHMENT 3

RE-TYPED CATAWBA
TECHNICAL SPECIFICATION

Amendment Number	Additional Condition		Implementation Date
	• The 72 hour allowed outage time of Technical Specification 3.7.5 Action "B" for the 1B AFW pump which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008. • The 72 hour allowed outage time of Technical Specification 3.6.6 Action "A" for the 1B CSS which was entered at 1041 on July 12, 2008 may be extended by an additional 144 hours. Upon completion of the repair and restoration of the 1B NSWS pump, this License Condition is no longer applicable and will expire at 1041 on July 21, 2008.		July 15, 2008 at 1041

ATTACHMENT 4

REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS

The following list identifies those actions committed to by Catawba in this document, for the duration of the extension. Any other statements made in this licensing submittal are provide for informational purposes only and are not considered to be regulatory commitments. Please direct any questions you may have in this matter to A.P. Jackson at (803) 701-3742.

Regulatory Commitment	Due Date
The proposed changes to the Catawba TS will be implemented prior to the end of the original 72 hour Completion Time (1041 on 7/15/08).	July 15, 2008
During the extended Completion Time period, no major maintenance or testing will be planned on the remaining operable NSWS "A" header. In addition, for Unit 1, during this period, no major maintenance or testing will be planned on the operable equipment that relies upon "A" Train NSWS as a support system.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on the Unit 1A AFW pump and Unit 1 turbine driven AFW pump.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on the Unit 1 A train and B train CCW system.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on the Unit 1 A train Chemical Volume and Control System.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on the SSF.	July 15, 2008
No major maintenance or testing will be planned on the portions of the drinking water system that are relied upon to provide backup cooling to the "A" charging pumps.	July 15, 2008
During the extended Completion Time period, the AFW system train 1B motor driven will remain available.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on the 1A and 1B essential AC power switchgear including 4160 volt busses, load centers and motor control centers.	July 15, 2008
During this extended Completion Time period, no major maintenance or testing will be planned on switchyard components, the 1A and 2A emergency diesel generators, and the transformers that feed the 1A, 1B, and 2A 4160 volt busses.	July 15, 2008
An action taken by Catawba to reduce the likelihood of an operator failing to get to the SSF and performing the required actions is to station an individual in the SSF continuously. This individual is trained on how to operate the SSF diesel generator and the standby makeup pump to establish an alternate method of reactor coolant pump seal injection. This will provide additional assurance that the SSF will be available during the extended completion time.	July 15, 2008
Prior to entering the extended Completion Time the operating crews will review the procedures regarding starting the SSF and establishing backup cooling to an "A" charging pump.	July 15, 2008
Catawba will perform a cleanliness inspection while the NSWS pit B is drained for pump change out.	July 21, 2008
To mitigate the risk of a potential core damage event, an operator action has been identified. This involves dispatching operators to throttle key AFW valves to supply the flow to the steam generators prior to the depletion of the vital batteries, thereby preventing steam	July 15, 2008

generator overfill and thus protecting the steam supplies to the AFW turbine driven pump. Catawba will dedicate an operator on each shift with this responsibility.	
Catawba has installed permanent flood protection barriers in the turbine building to mitigate turbine building flooding. In addition, to help reduce any potential flooding issues, no major maintenance or testing will be planned on the Condenser Circulating Water System.	July 15, 2008
Duke commits to a change out of the 1B NSWS pump at the next scheduled refueling outage.	Prior to the end of 1EOC 18 Refueling outage
Catawba will perform a detailed cause evaluation of this failure of the 1B NSWS pump. This cause evaluation will be compared to the cause evaluation completed for the failure of the 1B centrifugal charging pump earlier this year. This result of this comparison will be used to evaluate any potential enhancements to Catawba's pump preventive maintenance program.	November 30, 2008.

NOTES:

- The above commitments do not preclude the performance of any TS required surveillances provided that the surveillances do not render equipment inoperable.
- The above commitments do not preclude any work that becomes necessary due to emergent equipment issues. Any proposed work would be evaluated against this submittal and the appropriate risk mitigate measures would be put in place.

ATTACHMENT 5

CATAWBA PRA QUALITY DISCUSSION

ATTACHMENT 5

Table A-1
Catawba PRA – Open ASME PRA Standard Supporting Requirements

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
AS-A9	USE realistic, applicable (i.e., from similar plants) thermal hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems. (See SC-B4.)	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Perform analyses with the most up-to-date version of MAAP.	Technical	No impact is expected since success criteria are consistent with peer plants per the PWROG PSA Database.
AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Cut set review during model integration and when supporting applications should address this. Suggest adding this guidance to workplace procedure XSAA-103.	Technical	Many phenomenological effects are already considered in the model. Cut set review considers possible additional phenomenological effects.
DA-A1a	ESTABLISH definitions of SSC boundaries, failure modes, and success criteria consistent with corresponding basic event definitions in Systems Analysis (SY-A5, SY-A7, SY-A8, SY-A10 through SY-A13 and SY-B4) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events consistent with corresponding definitions in Systems Analysis (SY-A18).	No	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to discuss component boundaries definitions.	Documentation	No impact is expected for documentation issues.

ATTACHMENT 5

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
DA-B1	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage to the extent supported by data: (a) mission type (e.g., standby, operating) (b) service condition (e.g., clean vs. untreated water, air)	Partial	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to segregate standby and operating component data. Segregate components by service condition to the extent supported by the data.	Technical	This is a refinement to the equipment failure rates. However, since most components are grouped appropriately, the overall impact should be small.
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently)	Partial	Catawba Failure Rate Database, CNC-1535.00-00-0029, Rev. 2, January 2006	Revise the data calc. to include a specific discussion of outlier treatment (i.e., do any outliers exist? If so, how are these events considered and grouped?)	Documentation	No impact is expected for documentation issues.
DA-C13	EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) based on actual plant experience. CALCULATE coincident maintenance unavailabilities that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares," i.e., plant systems that have more redundancy than is addressed by tech specs. For example, the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with tech specs.	Partial	Developing PRA Data, Workplace Procedure XSAA-110, Rev. 4, July 2007; Catawba Component Failure Rate Denominator Estimates, SAAG 492, December 1997	Put in place a mechanism for identifying and quantifying coincident unavailabilities. Incorporate in the system models those maintenance events allowed by technical specifications where 2 or more components have maintenance events that are correlated with each other.	Technical	No impact since additional maintenance on important equipment will be restricted during the amended completion time.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
DA-D4	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Partial	Catawba PRA Common Cause Analysis, CNC-1535.00-00-0028, Rev. 0, December 2005	Enhance the documentation to include a discussion of the specific checks performed on the Bayesian-updated data, as required by this SR.	Documentation	No impact is expected for documentation issues.
DA-D6	USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities consistent with the component boundaries.	Partial	Catawba PRA Common Cause Analysis, CNC-1535.00-00-0028, Rev. 0, December 2005	Provide documentation in SAAG 637 of the comparison of the component boundaries assumed for the generic CCF estimates to those assumed in the Catawba PRA to ensure that these boundaries are consistent.	Documentation	No impact is expected for documentation issues.

ATTACHMENT 5

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-A2	IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Enhance the HRA to consider the potential for calibration errors.	Technical	Based on preliminary evaluations using the EPRI HRA calculator, calibration errors that result in failure of a single channel are expected to fall in the low 10^{-3} range. Calibration errors that result in failure of multiple channels are expected to fall in the low 10^{-5} range. Relative to post-initiator HEPs, equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-A3	IDENTIFY which of those work practices identified above (HR-A1, HR-A2) involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems [e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system (e.g., SLCS)].	No		This is the documentation part of the issue described in SR's HR-A1 and HR-A2.	Documentation	No impact is expected for documentation issues.
HR-D6	PROVIDE an assessment of the uncertainty in the HEPs. USE mean values when providing point estimates of HEPs.	No		Develop mean values for pre-initiator HEPs.	Technical	Pre-initiator HEPs are generally set to relatively high screening values. Thus the suggested data refinement is not expected to have a significant impact on this application.
HR-G3	When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of the meaning of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of detection, diagnosis and decision-making, and executing the required response (h) environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Document in more detail the influence of performance shaping factors on execution human error probabilities.	Documentation	No impact is expected for documentation issues.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation?	Expected Impact on Application
HR-G4	BASE the time available to complete actions on appropriate realistic generic thermal-hydraulic analyses, or simulation from similar plants (e.g., plant of similar design and operation) (See SC-B4.). SPECIFY the point in time at which operators are expected to receive relevant indications.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Enhance HRA documentation accordingly.	Documentation	No impact is expected for documentation issues.
HR-G6	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	No		Document a review of the HFEs and their final HEPs relative to each other to confirm their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	Documentation	No impact is expected for documentation issues.
HR-G9	Characterize the uncertainty in the estimates of the HEPs, and PROVIDE mean values for use in the quantification of the PRA results.	No		Develop mean values for post-initiator HEPs.	Technical	Use of mean values for HEPs is expected to result in an increase in post-initiator HEP values, in the base case model as well as for applications. Implementing compensatory actions for the important operator actions is expected to have an offsetting effect, thereby reducing the HEPs.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis: (a) a procedure is available and operator training has included the action as part of crew's training, or justification for the omission for one or both is provided (b) "cues" (e.g., alarms) that alert the operator to the recovery action provided procedure, training, or skill of the craft exist (c) attention is given to the relevant performance shaping factors provided in HR-G3 (d) there is sufficient manpower to perform the action.	Partial	Catawba Human Reliability Analysis, CNC-1535.00-00-0030, Rev. 0, December 2005	Develop more detailed documentation of operator cues, relevant performance shaping factors, and availability of sufficient manpower to perform the action.	Documentation	No impact is expected for documentation issues.
IE-A1	IDENTIFY those initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage using a structured, systematic process for identifying initiating events that accounts for plant-specific features. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA). Existing lists of known initiators are also commonly employed as a starting point.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006; Systems Analysis	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A3	REVIEW the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience. See also IE-A7	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Perform a review of the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience.	Technical	Initiating events (other than ATWS) result in a plant trip and the generation of an LER. These events are reviewed as part of the initiating events analysis. Fire and flood events that don't result in a reactor trip could potentially impact the frequencies assigned to the fire and flood initiators. However, fire and flood sequences are not significant contributors to the delta CDF in the PRA analysis for the LAR. Thus this open SR does not have a significant impact.
IE-A3a	REVIEW generic analyses of similar plants to assess whether the list of challenges included in the model accounts for industry experience.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Ensure the list of challenges included in the Catawba PRA accounts for industry experience using a more recent reference, such as the WOG PSA Model and Results Comparison Database - Revision 4.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref:	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A4	PERFORM a systematic evaluation of each system where necessary (e.g., down to the subsystem or train level), including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE a structured approach [such as a system-by-system review of initiating event potential, or an FMEA (failure modes and effects analysis), or other systematic process] to assess and document the possibility of an initiating event resulting from individual systems or train failures.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Provide documentation of a systematic evaluation of all plant systems, including support systems (including those not explicitly modeled in the PRA), to assess the possibility of an initiating event occurring due to a failure of the system.	Documentation	No impact is expected for documentation issues.
IE-A4a	When performing the systematic evaluation required in IE-A4, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from system alignments resulting from preventive and corrective maintenance.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-A5	In the identification of the initiating events, INCORPORATE (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation. (b) events resulting in a controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-A6	INTERVIEW plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	No		Obtain plant personnel input (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IE-A7	REVIEW plant-specific operating experience for initiating event precursors, for the purpose of identifying additional initiating events. For example, plant specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Include review of precursor events for their potential to be initiating events.	Documentation	No impact is expected for documentation issues.
IE-B1	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis element (para. 4.5.2) and to facilitate quantification in the Quantification element (para. 4.5.8).	No		Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-B2	USE a structured, systematic process for grouping initiating events. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA).	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Document a structured, systematic grouping of initiating events (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation?	Expected Impact on Application
IE-B3	GROUP initiating events only when the following can be assured: (a) events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or (b) events can be subsumed into a group and bounded by the worst case impacts within the "new" group. DO NOT SUBSUME events into a group unless: (1) the impacts are comparable to or less than those of the remaining events in that group, AND (2) it is demonstrated that such grouping does not impact significant accident sequences.	Partial	Catawba Internal Initiator Event Frequency Data, CNC-1535.00-00-0031, Rev. 0, January 2006	Enhance documentation of the grouping process (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IE-D3	DOCUMENT the assumptions and sources uncertainty with the initiating event analysis.	No		Enhance the IE documentation (as was done in OSC-9068).	Documentation	No impact is expected for documentation issues.
IF-B3	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE: (a) a characterization of the breach, including type (e.g., leak, rupture, spray) (b) range of flow rates (c) capacity of source (e.g., gallons of water) (d) the pressure and temperature of the source	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Enhance the Internal Flood analysis to address the potential for spray, jet impingement, and pipe whip failures. Additionally, document how these failures are included in the quantification.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-C2c	For each flood area not screened out using the requirements under IF-B1b, IDENTIFY the SSCs located in each defined flood area and IF-A2) along flood propagation paths that are modeled in the internal events PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibility per IF-C3, its spatial location in the area and any flooding mitigative features (e.g., shielding, flood or spray capability ratings).	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	For those flood areas addressed in the current flooding analysis, equipment important to accident mitigation and the associated critical flood heights are identified. However, given the expected increase in number of flood areas needed to satisfy requirement IF-A1, additional equipment will need to be identified and discussed in order to meet the requirements of the ASME Standard. The current flooding analysis does not discuss flood mitigative features and this will have to be corrected to satisfy the requirements of the ASME Standard.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
IF-C3	For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	The current flooding analysis identifies the submergence failure height of the equipment important to accident mitigation, but never addresses the impact of spray. Spray as a failure mechanism needs to be addressed in the analysis or a note made explaining why it was omitted.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-C3b	IDENTIFY inter-area propagation 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. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads and the potential for barrier unavailability, including maintenance activities.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Provide more analysis of flood propagation flowpaths. Address potential structural failure of doors or walls due to flooding loads and the potential for barrier unavailability.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.
IF-E6b	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and indirect effects such as submergence, jet impingement, and pipe whip, as applicable.	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Address potential indirect effects.	Technical	Internal flood sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
IF-F2	DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification. For example, this documentation typically includes (a) flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined (b) flood areas used in the analysis and the reason for eliminating areas from further analysis (c) propagation pathways between flood areas and assumptions, calculations, or other bases for eliminating or justifying propagation pathways (d) accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification (e) assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability (f) screening criteria used in the analysis (g) flooding scenarios considered, screened, and retained (h) description of how the internal event analysis models were modified to model these remaining internal flooding scenarios (i) flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) (j) calculations or other analyses used to support or refine the flooding evaluation (k) results of the internal flooding analysis, consistent with the quantification requirements provided in HLR QU-D	Partial	Catawba Flood Analysis, CNC-1535.00-00-0058, Rev. 0, December 2005	Need to document how the analysis addressed all of the items identified in this requirement.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
LE-C10	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Perform plant-specific T/H calculations for SGTR. Consider some credit for ISLOCA scrubbing; if no credit can be given, then this should be documented. It is not known whether or not the additional analysis will alter the LERF, but because these items dominate LERF, a more realistic analysis should be considered.	Technical	The conservative treatment will not mask the contribution of non-bypass events, because even if some credit were given to scrubbing, the unscrubbed bypasses would still dominate LERF over the non-bypass events. In addition, the limiting risk metric in the present analysis is CDF, not LERF.
LE-C6	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of para. 4.5.5, as appropriate for the level of detail of the analysis.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	The only operator action expected to be important is RCS depressurization for small LOCAs. However, the current analysis lacks a formal dependency analysis for this action. The result is expected to be insensitive to this impact given that the SGTR so dominates the result.	Documentation	No impact is expected for documentation issues.
LE-D3	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for non-significant accident progression sequences resulting in a large early release. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	No	Catawba ISLOCA Analysis, CNC-1535.00-00-0053, Rev. 0, January 2006	For MNS/CNS, the ND heat exchanger is assumed to provide the largest break flow area. The ISLOCA is a dominant contributor and the evaluation is relatively conservative.	Technical	ISLOCA sequences are not significant contributors in the present analysis. No significant impact associated with this open SR.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation?	Expected Impact on Application
LE-F2	PROVIDE uncertainty analysis that identifies the sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004; Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	Documentation	No impact is expected for documentation issues.
LE-F3	IDENTIFY contributors to LERF and characterize LERF uncertainties consistent with the applicable requirements of Tables 4.5.8-2(d) and 4.5.8-2(e). NOTE: The supporting requirements in these tables are written in CDF language. Under this requirement, the applicable requirements of Table 4.5.8 should be interpreted based on LERF, including characterizing key modeling uncertainties associated with the applicable contributors from Table 4.5.9-3. For example, supporting requirement QU-D5 addresses the significant contributors to CDF. Under this requirement, the contributors would be identified based on their contribution to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Compare LERF results and uncertainties to similar plants and include in the LERF documentation.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation?	Expected Impact on Application
LE-G3	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Evaluate the relative contribution of the various contributors to the total LERF.	Documentation	No impact is expected for documentation issues.
LE-G4	DOCUMENT assumptions and sources of uncertainty associated with the LERF analysis, including results and important insights from sensitivity studies.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	Documentation	No impact is expected for documentation issues.
LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	No		Include in the LERF documentation an assessment that identifies the limitations in the LERF analysis that could impact applications.	Documentation	No impact is expected for documentation issues.
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Partial	Catawba Simplified LERF Methodology, SAAG 817, Rev. 1, October 2004	Provide a discussion of the significant cut sets and sequences.	Documentation	No impact is expected for documentation issues.
QU-D3	COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another?	No		Perform and document a comparison of results between the CNS PRA and other similar plants.	Documentation	No impact is expected for documentation issues.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
QU-E4	EVALUATE the sensitivity of the results to model uncertainties and assumptions using sensitivity analyses [Note (1)].	No		Perform and document a set of sensitivity cases to determine the impact of the assumptions and sources of model uncertainty on the results.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
QU-F2	DOCUMENT the model integration process, including any recovery analysis, and the results of the quantification including uncertainty and sensitivity analyses. For example, documentation typically includes (a) records of the process/results when adding nonrecovery terms as part of the final quantification (b) records of the cutset review process (c) a general description of the quantification process including accounting for systems successes, the truncation values used, how recovery and post-initiator HFEs are applied (d) the process and results for establishing the truncation screening values for final quantification demonstrating that convergence towards a stable result was achieved (e) the total plant CDF and contributions from the different initiating events and accident classes (f) the accident sequences and their contributing cutsets (g) equipment or human actions that are the key factors in causing the accident sequences to be nonsignificant (h) the results of all sensitivity studies (i) the uncertainty distribution for the total CDF (j) importance measure results (k) a list of mutually exclusive events eliminated from the resulting cutsets and their bases for Elimination (l) asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model (m) the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Expand the documentation of CNS PRA model results to address all required items.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
QU-F6	DOCUMENT the quantitative definition used for significant basic event, significant cutset, significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	Partial	Catawba Rev 3a PRA Model Integration Notebook, CNC-1535.00-00-0061, Rev. 2, July 2006	Document the required definitions.	Documentation	No impact is expected for documentation issues.
SC-A4	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.
SC-B5	CHECK the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include: (a) comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features (b) comparison with results of similar analyses performed with other plant-specific codes (c) check by other means appropriate to the particular analysis	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Provide evidence that an acceptability review of the T/H analyses is performed.	Documentation	No impact is expected for documentation issues.
SC-C1	DOCUMENT the success criteria in a manner that facilitates PRA applications, upgrades, and peer review.	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SC-C2	DOCUMENT the processes used to develop overall PRA success criteria and the supporting engineering bases, including the inputs, methods, and results. For example, this documentation typically includes: (a) the definition of core damage used in the PRA including the bases for any selected parameter value used in the definition (e.g., peak cladding temperature or reactor vessel level) (b) calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used (c) identification of computer codes or other methods used to establish plant-specific success criteria (d) a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) of the calculations or codes (e) the uses of expert judgment within the PRA, and rationale for such uses (f) a summary of success criteria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA (g) the basis for establishing the time available for human actions (h) descriptions of processes used to define success criteria for grouped initiating events or accident sequences	Partial	Catawba Thermal-Hydraulic Success Criteria calcs.	Improve the documentation on the TH bases for all safety function success criteria for all initiators.	Documentation	No impact is expected for documentation issues.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-A14	In meeting SY-A12 and SY-A13, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met: (a) A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. (b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same.	Partial	System analyses	Provide quantitative evaluations for screening.	Documentation	No impact is expected for documentation issues.
SY-A2	COLLECT pertinent information to ensure that the systems analysis appropriately reflects the as-built and as-operated systems. Examples of such information include system P&IDs, one-line diagrams, instrumentation and control drawings, spatial layout drawings, system operating procedures, abnormal operating procedures, emergency procedures, success criteria calculations, the final or updated SAR, Technical Specifications, training information, system descriptions and related design documents, actual system operating experience, and interviews with system engineers and operators.	Partial	System analyses	Need to update references per XSAA-115.	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-A4	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	Partial	System analyses	Enhance the system documentation to include an up-to-date system walkdown checklist and system engineer review for each system. Consider revising workplace procedure XSAA-106 to require that such documentation be revisited with each major PRA revision.	Documentation	No impact is expected for documentation issues.
SY-A8	ESTABLISH the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data. For example, a control circuit for a pump does not need to be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL as separate basic events of the model, those subcomponents (e.g., a valve limit switch that is associated with a permissive signal for another component) that are shared by another component or affect another component, in order to account for the dependent failure mechanism.	No		Enhance systems analysis documentation to discuss component boundaries.	Documentation	No impact is expected for documentation issues.

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SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
SY-B15	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include: (a) LOCA inside containment with failure of containment heat removal (b) safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) (c) steam line breaks outside containment (d) debris that could plug screens/filters (both internal and external to the plant) (e) heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability (f) loss of NPSH for pumps (g) steam binding of pumps (h) harsh environments induced by containment venting or failure that may occur prior to the onset of core damage	Partial	System analyses	Cut set review during applications should address this. Suggest adding this guidance to workplace procedure XSAA-103.	Documentation	No impact is expected for documentation issues.
SY-B8	IDENTIFY spatial and environmental hazards that may impact multiple systems or redundant components in the same system, and ACCOUNT for them in the system fault tree or the accident sequence evaluation. Example: Use results of plant walkdowns as a source of information regarding spatial/environmental hazards, for resolution of spatial/environmental issues, or evaluation of the impacts of such hazards.	Partial	System analyses	Per Duke's PRA modeling guidelines, ensure that a walkdown/system engineer interview checklist is included in each system notebook. Based on the results of the system walkdown, summarize in the system write-up any possible spatial dependencies or environmental hazards that may impact system operation.	Documentation	No impact is expected for documentation issues.
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human	Partial	System analyses	Enhance system model documentation to comply with all ASME PRA	Documentation	No impact is expected for documentation issues.

SR	Category II Requirements	Met for CNS?	CNS Ref.	Resolution	Technical or Documentation ?	Expected Impact on Application
	actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and results. For example, this documentation typically includes: (a) system function and operation under normal and emergency operations (b) system model boundary (c) system schematic illustrating all equipment and components necessary for system operation (d) information and calculations to support equipment operability considerations and assumptions (e) actual operational history indicating any past problems in the system operation (f) system success criteria and relationship to accident sequence models (g) human actions necessary for operation of system (h) reference to system-related test and maintenance procedures (i) system dependencies and shared component interface (j) component spatial information (k) assumptions or simplifications made in development of the system models (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes (m) a description of the modularization process (if used) (n) records of resolution of logic loops developed during fault tree linking (if used) (o) results of the system model evaluations (p) results of sensitivity studies (if used) (q) the sources of the above information (e.g., completed checklist from walkdowns, notes from discussions with plant personnel) (r) basic events in the system fault trees so that they are traceable to modules and to cutsets. (s) the nomenclature used in the system models.			Standard requirements.		