Enclosure 6

Case Study 2: Catawba Emergency TS Change

Meeting Summary of the January 27 & 28 Meeting with NRC/TSTF

Dated March 9, 2009

Day 1

Case Study 2: Catawba Emergency TS Change

On July 12, 2008, at approximately 1041 hours, the 1B Nuclear Service Water System (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/O/A/6400/006 C, "Nuclear Service Water System, Enclosure 4.12B." This procedure isolates NSWS flow to the 1B AFW pump, the 1B Containment Spray System (CSS) heat exchanger and the Unit 1 nonessential NSWS header. The licensee believed that this allowed the "B" NSWS train to be declared operable and both units exited TS 3.7.8. The staff does not agree with the licensee's position. The realignment of the system is discussed in the bases, but is not mentioned in the TS 3.7.8. TS 3.7.8 requires to operable trains of NSWS. With one train inoperable, the TS require the inoperable train to be restored to operable within 72 hours.

Discussion

ISTS 3.7.8 requires two SWS trains shall be OPERABLE. The ISTS LCO Bases state:

Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An SWS train is considered OPERABLE during MODES 1, 2, 3, and 4

when:

a. The pump is OPERABLE and

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

In practice, however, most plants description of an Operable SWS is more complex. Especially for two unit sites, there are redundant pumps and piping arrangements which would allow the systems to " provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power." These alternate arrangement are described the LCO Bases.

During the development of the ISTS, the NRC suggested that the details of what constitutes Operability be moved from the LCO to the LCO Bases. For example, the pre-ISTS ECCS LCO 3.5.2, "ECCS - Operating" looked like this:

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump (four loop plants only),

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

In the ISTS, the LCO looks like this:

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

The detailed discussion of what constitutes an Operable ECCS subsystem is moved to the LCO Bases, where it is controlled under the Bases Control Program.

In addition, SR 3.7.8,1 is modified by a Note, Isolation of SWS flow to individual components does not render the SWS inoperable." As described in the Bases, "This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SWS."

That is precisely what was done at Catawba.

Therefore, there Catawba's actions appear to be reasonable and consistent with their TS.

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One SWS train inoperable.	A.1	--------------NOTES------------- 1. Enter applicable and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SWS. ------------------------------------ Restore SWS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.8.1	-------------------------------NOTE------------------------------ Isolation of SWS flow to individual components does not render the SWS inoperable. --- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	[18] months
SR 3.7.8.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	[18] months

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SW loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW loop is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Either R10 R13

a.1 Two SW pumps are OPERABLE in an OPERABLE flow path; or

a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled; and R13

b. Three spray arrays are OPERABLE in an OPERABLE flow path; and R10

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

For two SW loops to be considered OPERABLE during MODES 1, 2, 3, and 4, the following conditions must also be met in order to provide protection for a single active failure of the actuation circuitry: R10

a. With one SW pump operating on each SW loop, the operating pumps have opposite train designations; and

b. With one of the four spray arrays on each SW loop inoperable, the inoperable spray arrays have opposite train designations.

A required valve directing flow to a spray array, bypass line, or other component is considered OPERABLE if it is capable of automatically moving to its safety position or if it is administratively placed in its safety position.

Duke Energy® 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.

A001
NRR

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

James R. Morris

Attachments

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. Current Requirements :

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. Basis for Current Requirements:

LCO 3.7.5 Basis Discussion

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.

LCO 3.6.6 Basis Discussion

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