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May 3, 2005

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

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specification Amendment
Technical Specification 3.5.2, Emergency Core Cooling System, 3.6.6, Containment Spray System, 3.6.17, Containment Valve Injection Water System, 3.7.5, Auxiliary Feedwater System, 3.7.7, Component Cooling Water System, 3.7.8, Nuclear Service Water System, 3.7.10, Control Room Area Ventilation System, 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System, & 3.8.1, AC Sources - Operating

Reference: 1) Letter from Dhiaa Jamil to U.S. Nuclear Regulatory Commission dated November 16, 2004.

2) Letter from Sean Peters, U.S. Nuclear Regulatory Commission, to Duke Energy Corporation dated February 16, 2005.

Pursuant to 10 CFR 50.90, Duke Energy Corporation is submitting a revision to an amendment request submitted on November 16, 2004 to the Catawba Nuclear Station Facility Operating License and Technical Specifications (TS). The proposed TS changes will allow the "A" and "B" Nuclear Service Water System (NSWS) headers for each unit to be taken out of service for up to 14 days each for system upgrades.

This revision is based on discussions with the NRC during a meeting held on January 31, 2005 and the meeting summary documented in reference 2. The meeting was productive and Catawba has revised the amendment package based on the results.

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Reference 2 included some additional questions from the NRC staff that were not discussed in the January 31, 2005 meeting. The revisions are in sections 3.0 and 4.0 of Enclosure 1.0. For completeness, the entire Enclosure 1.0 is attached. Attachment 1 contains Catawba's response to the NRC requests documented in reference 2. Attachment 2 provides a summary of regulatory commitments made in this submittal.

The conclusions reached in the original determination that the amendment contains No Significant Hazards Considerations pursuant to 10 CFR 50.92, and the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement pursuant to 10 CFR 51.22(c)(9) have not been changed based on the revisions in the attachment to this letter.

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

Inquiries on this matter should be directed to R. D. Hart at (803) 831-3622.

Very truly yours,

Dhiaa Jamil

RDH/s

Enclosure: 1) - EVALUATION
Attachments: 1) - ANSWERS TO NRC REQUESTS IN 2/16/05 LETTER
2) - SUMMARY OF REGULATORY COMMITMENTS
3) - APPLICABLE PEER REVIEW ITEMS

Dhiaa Jamil 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.

Dhiaa Jamil, Site Vice President

Subscribed and sworn to me: 05-03-05
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

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ENCLOSURE 1

EVALUATION

EVALUATION

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy requests one-time temporary changes to Technical Specification (TS) 3.5.2, Emergency Core Cooling System (ECCS) - Operating, 3.6.6, Containment Spray System (CSS), 3.6.17, Containment Valve Injection Water System (CVIWS), 3.7.5, Auxiliary Feedwater (AFW) System, 3.7.7, Component Cooling Water (CCW) System, 3.7.8, Nuclear Service Water System (NSWS), 3.7.10, Control Room Area Ventilation System (CRAVS), 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), and 3.8.1 AC Sources - Operating for Catawba Nuclear Station Units 1 and 2. The proposed TS changes will allow the "A" & "B" Nuclear Service Water System (NSWS) headers for each unit to be taken out of service for up to 14 days to allow coating of weld(s) associated with the piping. This will be a one-time evolution for each header. This evolution is scheduled to occur when Unit 1 and 2 are at power operation. The references cited in this amendment are listed in section 7.0.

2.0 Proposed Changes

Duke Energy proposes to temporarily change TS 3.5.2, ECCS - Operating, 3.6.6, Containment Spray System, 3.6.17, Containment Valve Injection Water System, 3.7.5, Auxiliary Feedwater (AFW) System, TS 3.7.7, Component Cooling Water System, 3.7.8, Nuclear Service Water System, TS 3.7.10, Control Room Area Ventilation System (CRAVS), TS 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), and 3.8.1 AC Sources - Operating to allow operation of the NSWS with one train inoperable on both units for one time period of up to 14 days for each NSWS train.

<u>TS 3.5.2, "ECCS - Operating"</u>

The following footnote will be added for the ECCS system to temporarily allow one train of ECCS to be inoperable for up to 14 days:

> *For each Unit, the Completion Time that one ECCS train can be inoperable as specified by Required Action A.1 may be extended beyond the 72 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades

and system restoration, this footnote is no longer applicable.

TS 3.6.6, "Containment Spray System"

The following footnote will be added for the Containment Spray System to temporarily allow one train of containment spray to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one Containment Spray System train can be inoperable as specified by Required Action A.1 may be extended beyond the 72 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.6.17 "Containment Valve Injection Water System (CVIWS)"

The following footnote will be added for the CVIWS to temporarily allow one train of CVIWS to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one CVIWS train can be inoperable as specified by Required Action A.1 may be extended beyond the 168 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.7.5 "Auxiliary Feedwater (AFW) System"

The following footnote will be added for the AFW system to temporarily allow one train of AFW to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one AFW train can be inoperable as specified by Required Action B.1 may be extended beyond the "72 hours and 10 days from discovery of failure to meet the LCO" up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.7.7 "Component Cooling Water (CCW) System

The following footnote will be added for the CCW system to temporarily allow one train of CCW to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one CCW train can be inoperable as specified by Required Action A.1 may be extended beyond the 72 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.7.8 "Nuclear Service Water System"

The following footnote will be added for the NSWS to temporarily allow one train of NSWS to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one NSWS train can be inoperable as specified by Required Action A.1 may be extended beyond the 72 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.7.10 "Control Room Area Ventilation System"

The following footnote will be added for the CRAVS to temporarily allow one train of CRAVS to be inoperable for up to 14 days:

*For each CRAVS train, the Completion Time that one CRAVS train can be inoperable as specified by Required Action A.1 may be extended beyond the 168 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.7.12 "Auxiliary Building Filtered Ventilation Exhaust System"

The following footnote will be added for the ABFVES to temporarily allow one train of ABFVES to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one ABFVES train can be inoperable as specified by Required Action A.1 may be extended beyond the 168 hours up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

TS 3.8.1 "AC Sources - Operating

The following footnote will be added for the EDGs to temporarily allow one train of NSWS to be inoperable for up to 14 days:

*For each Unit, the Completion Time that one EDG can be inoperable as specified by Required Action B.4 may be extended beyond the "72 hours and 6 days from discovery of failure to meet the LCO" up to 336 hours as part of the NSWS system upgrades. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Upon completion of the system upgrades and system restoration, this footnote is no longer applicable.

Since these changes in the TS are one-time changes, the associated TS Bases do not require any revision.

3.0 Background

On October 4, 2000 the NRC issued a TS amendment for the Catawba Nuclear Station to temporarily revise several TS sections to allow those systems to be inoperable for up to 12 days for NSWS system upgrades. The upgrades included a cleaning and pipe replacement project that was completed during the Unit 1 refueling outage in the fall of 2000. The cleaning, pipe replacement, and testing were performed in 9 ¼ days for train A and 8 ¼ days for B train of the NSWS. This was well within the time frame of 12 days granted by the previous license amendment. The work was performed safely and no licensee event reports (LERs) were generated as a result of this work. This project allowed the inspection of intake structures, cleaning of the NSWS pump house, and cleaning of approximately 8000 linear feet of NSWS piping in various sizes. The cleaning process removed corrosion products, silt, sediment, and biological build-up from the pipe inside diameter and cleaned the pipe to almost bare metal. The cleaning also allowed for an internal inspection of the NSWS piping. This inspection included visual, UT and video taping to document the condition of the NSWS piping after cleaning. Remote cameras were used to videotape internal sections of the piping.

The results of these inspections have been documented in the Catawba corrective action program for review and disposition. This has resulted in identifying the most limiting portions of NSWS piping to schedule repairs and/or replacement.

Based on the pipe inspections from the 2000 cleaning, a 20 foot section of NSWS piping was targeted for replacement during a 7 day LCO in January of 2003. This 7 day LCO was granted by the NRC via license amendment 203 and 196 dated January 7, 2003. This section of piping was selected for replacement due to the longitudinal seam weld being located in the bottom of the pipe below the silt and sediment layer that was removed from the piping. Based on the inspections, this weld was determined to be the worst case available for examination and testing. This section of piping was subjected to extensive examination and successfully hydrostatically tested to 150% of design pressure. Corrosion at the longitudinal seam weld in the NSWS pipe section occurred in both the heat-affected zone of the base metal and in the weld filler metal itself. Corrosion occurred more readily in the heat-affected zone/fusion zone area, which led to the formation of grooves along the length of the longitudinal seam weld. The Metallurgical Lab report on the seam welds provides the following information:

> "Both the welds and heat-affected zones were subject to corrosion, while the base metal was not. This type of preferential corrosion in carbon steel welds has been observed in other applications but cannot be precisely explained. The difference in corrosion potential may be due to:
> a) Compositional differences, in which the weld/HAZ is anodic to the base metal;
> b) Different amounts of entrained inclusions/deoxidation products;
> c) Variations in corrosion behavior of different steel microstructures, particularly in slightly acidic conditions which may have existed beneath the sludge layer;
> d) Some combination of these three factors.
>
> The sludge layer that was present atop the seam weld at the pipe bottom almost certainly increased the aggressiveness of the environment. MIC activity may have decreased the pH level of the environment beneath the sludge layer. Seam welds positioned elsewhere around the pipe circumference would be expected to corrode more slowly than the seam weld examined here, despite the probability that they may also be anodic to the base metal."

Corrosion removed both the weld filler metal and the heat-affected zone of the base metal, but left unaffected base metal intact. Areas of corrosion in the circumferential welds were also observed. The attack did not penetrate as deeply as did that in the longitudinal seam weld due to differences in weld geometry.

A review of the degraded condition was performed and documented in the Catawba corrective action program. The operability of the removed section was justified by the hydrostatic test. This successful test demonstrated the ability of the degrading piping to withstand significantly higher pressure loadings than those that would be encountered during any normal or accident condition. The hydrostatic test was chosen to justify operability since the wall thickness of the NSWS buried piping required to meet ASME Code requirements is controlled by internal pressure. With regards to seismic loads, the passing of a seismic wave in the soil results in bending, compressive, and tensile strains in the buried piping. Flexural strains are typically negligible for normal pipe diameters, and so wave propagations are generally considered in terms of their impact on longitudinal axial strain, i.e. strain parallel to the pipe axis induced by the ground strain. These axial strains are displacement controlled as the piping moves with the soil and are not critical with respect to the pipe wall thickness as demonstrated in supporting calculations performed by Duke. Thus, the hydrostatic pressure test represents the critical loading with respect to the loss of pipe wall thickness. Also, from Duke's visual observations and conclusions from supporting calculations, the longitudinal seam weld was determined as critical with respect to structural integrity. The hoop stress developed during the hydrostatic test directly challenges the longitudinal seam weld.

With respect to the remaining parts of the NSWS system, several other bottom oriented seam welds were chosen for inspection based on the video. These inspections included two longitudinal seams and two circumferential welds in 2003, four longitudinal seam welds and portions of 4 circumferential welds in 2004. Two longitudinal seam welds inspected in 2003 were included in the four welds inspected in 2004. These welds were re-inspected to gage the rate of degradation post cleaning. These inspections did not reveal any minimum wall violations or significant change in the degradation between the 2003 and 2004 inspections. Also, no other conditions were identified that were not bounded by the cases within the cut out section of piping.

Based on the inspections performed, the welds in the NSWS supply header piping should be restored prior to 1EOC17, which corresponds to the spring of 2008. The initial activities include cleaning, inspection, and coating of NSWS piping welds, and based on the inspections may also include any necessary system repairs, replacement, or modifications and has been conservatively scheduled for the fall of 2005 or the first quarter of 2006.

3.1 Nuclear Service Water System Improvement Plan

Over the next several years, Catawba will implement a NSWS Improvement Plan that will lead to a more reliable NSWS. As each phase of this plan is implemented, the reliability of the NSWS and those safety systems that it supplies cooling water to will be improved. When complete, the NSWS piping will be in a physical state that will allow the station to operate with minimal impact to nuclear safety due to service water reliability or unavailability.

The NSWS plan is divided into three distinct phases. The initial phase of the plan specifically targets the stabilization of the welds in the NSWS supply headers. An intermediate phase of the plan is to implement a series of modifications and system enhancements which will restore the system to its original design and provide operational flexibility to allow for system maintenance with minimal impact to safety system availability. The intermediate phase will be scheduled to be completed within the existing TS time frames to the extent practical. During detailed review and planning it may become necessary to request additional TS changes to support some of this work. The final phase of the plan will be the coating, and any necessary repairs, of the NSWS supply header. This phase may be expanded to include repairs of the lake and Standby Nuclear Service Water Pond (SNSWP) return lines depending upon the results of the inspections that will be made in the initial phase of the plan.

To implement the final phase of this plan, an additional License Amendment Request will be needed to allow for the operation of both units from a single NSWS supply header. The cornerstone of this request will utilize a flow model of the NSW system which will accurately predict the flow rate and pressure of the various components in the NSWS system such that verification can be made that these components have sufficient flow and pressure to perform their design functions during single NSWS header operation.

The TS changes requested in section 2.0 of this enclosure will provide the time necessary to implement the first phase of this system upgrade project. System upgrades include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. Civil engineering evaluations of the longitudinal and circumferential welds in the supply headers have determined that the first priority area for the initial phase should be main buried 42 inch supply headers. These activities are being done to preclude any further degradation of the affected welds. This will allow the intermediate and final phases of the NSWS Improvement Plan to commence with a predictable and reliable schedule. The welds are currently scheduled to be repaired and/or coated in the fall of 2005 or first quarter of 2006 to help minimize the possibility of total replacement of the buried pipe in the final phase of the NSWS Improvement Plan.

An acceptable, immediate method of restoring these welds to a more serviceable condition is to provide an appropriate top coating to each weld to protect them from future degradation.

Performance of this maintenance activity during the proposed 14 day LCO period will allow for future long term maintenance activities on the supply headers to proceed with added assurance of the reliability of the in service header. By performing this activity, the short term reliability of the supply headers will be enhanced for all future maintenance activities which will address the long term reliability of the entire Nuclear Service Water System.

In parallel with the primary activity of cleaning, inspecting and coating NSWS piping welds, some parts of the intermediate phase will be implemented. These items will provide future flexibility for the implementation of the remainder of the intermediate and final phases of the plan such that minimal impact to safety system availability is realized. The portions of the intermediate phase are as follows and to the extent feasible will be planned for implementation during this 14 day period:

- Isolation valves will be installed on the discharge piping of each NSWS Pump (4 total) at the pump house wall to allow piping replacement and installation of pump house crossovers without affecting the operation of the other unit's train related pump.

- The existing supply headers to the Auxiliary Building will be modified inside the Auxiliary Building to allow piping replacement to be performed inside the Auxiliary Building with the supply header in service to the opposite unit.

- Isolation valves will be installed in each Unit's Emergency Diesel Generator Building's NSWS supply to allow the installation of crossover piping between the two trains of NSWS between the EDGs.

- Isolation valves will be installed on each side of the discharge crossovers to allow future piping replacement and coating to be implemented during refueling outages without impact to the operating unit.

The intermediate phase of the NSWS Improvement Plan involves an extensive series of modifications that will be implemented during future refueling outages and some non refueling outage periods. These modifications include enhancements which will allow for maintenance to be performed without significant impact to safety system availability and the replacement of existing carbon steel pipe with a more corrosion resistant material.

Catawba is planning future repair and upgrade activities that will require limited operation on a single NSWS supply header. Catawba will submit a license amendment to modify TS to allow limited operation on a single NSWS supply header. In this configuration, the NSWS headers will still maintain electrical train separation, but utilize a single supply header. Upon NRC approval of single header operation, each NSWS supply train will be removed from service with the appropriate valves aligned to provide cross train alignment.

The single header operation is partially based on engineering calculation CNC-1223.24-00-0027 (A Flow Distribution Model of the RN System). This calculation documents a hydraulic model of the NSWS system based on as built system piping isometric drawings. The PROTO-FLO software program is used to develop the calculation. This software has a flow balancing feature which allows the setting of throttle valve positions from actual system flow balances. With the model validated on the current plant configuration, a new calculation will be generated which revises the system based on the modifications which will have been performed during the intermediate phase of the refurbishment plan. This calculation will isolate one of the NSWS supply headers, open the Auxiliary Building train crossovers, align the NSWS Pump crossovers, and the Emergency Diesel Generator crossovers. The model will run per this alignment to prove that the single supply header aligned with either A or B train NSWS Pumps in operation can provide sufficient flow to all essential safety related components for all design basis events. To support this single header operation, a complete and thorough design study will be conducted to assure design basis events and possible operational situations have been identified and evaluated.

The proposed changes to TS requirements requested in section 2.0 of this enclosure for this license amendment provide the operational flexibility necessary to perform the activities associated with the first phase of a large project to enhance and ensure NSWS continued operation for the life of the plant. This phase includes activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. During the time period that one NSWS header is inoperable the opposite NSWS header and support systems will remain operable. This activity is based on recommendations from Engineering and the results of the video inspections and other analyses completed after the major system cleaning project completed in the fall of 2000.

3.2 Risk Informed Configuration Risk Management

3.2.1 Avoidance of Risk-Significant Plant Equipment Outage Configurations

Risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed License Amendment Request (LAR). Duke is not proposing any additional compensatory actions as a result of the proposed LAR.

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)," Revision 2, May 2004.
- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)," Revision 13, March 22, 2005.
- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL," Revision 8, June 2004.
- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL," Revision 7, June 2004.

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 considering the proposed LAR.

3.2.2 Maintenance Rule Configuration Control

10 CFR 50.65 (a)(4) (Ref. 16), RG 1.182 (Ref. 17), and NUMARC 93-01 (Ref. 18) 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 (Ref. 6) 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.

The key safety significant systems impacted by this LAR are currently included in the Maintenance Rule program, and as such, availability and reliability performance criteria have been established to assure that they perform adequately.

The large scope of this maintenance activity requires direct management involvement. Catawba Nuclear Station (CNS) Site Directive 3.0.18, "On-Line Maintenance," is the process to be used. This Site Directive is part of the overall configuration risk management program which is used to assess and manage risk from proposed maintenance activities. The structured approach in Site Directive 3.0.18 ensures the appropriate level of management attention throughout the project. It assures proper review, representation, and planning from appropriate on-site groups prior to execution of work. This process also provides step by step directions for the execution and completion of the project. Under the guidelines of this directive this project is considered a "Critical Maintenance Process" and will follow that format. The controlling document for the project is called the "Critical Maintenance Process Plan".

3.2.3 Probabilistic Risk Assessment (PRA) Model

PRA Model / Scope

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, (e.g., pumps, valves, and heat exchangers). There are no plant unit-specific differences that would impact the PRA model. This level of detail is sufficient for this application.

PRA Updates / Quality

Duke periodically evaluates changes to the plant with respect to the assumptions and modeling in the Catawba PRA. The original Catawba PRA was initiated in July 1984 by Duke Power Company assisted by several outside contractors who performed specialized subtasks. It was a full scope Level 3 PRA with internal and external events. A peer review sponsored by the Electric Power Research Institute (EPRI) was conducted after completion of the draft report. The study was published in an internal Duke report (Ref. 5) in 1987 as Revision 0 to the PRA.

On November 23, 1988, the NRC issued Generic Letter 88-20 (Ref. 7), which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. The Catawba response to GL 88-20 was provided by letter dated September 10, 1992 (Ref. 8). Catawba's response included an updated Catawba PRA (Revision 1) study.

The Catawba PRA Revision 1 study and the IPE process resulted in a comprehensive, systematic examination of Catawba with regard to potential severe accidents. The Catawba study was again a full-scope, Level 3 PRA with analysis of both the internal and external events. This examination identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on likelihood and fission product release potential. The results of the study prompted changes in equipment, plant configuration and enhancements in plant procedures to reduce vulnerability of the plant to some accident sequences of concern.

By letter dated June 7, 1994 (Ref. 9), the NRC provided a Safety Evaluation of the internal events portion of the above Catawba IPE submittal. The conclusion of the NRC letter (page 16) states:

> "The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal and the associated supporting information, the staff finds reasonable the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at Catawba."

In response to Generic Letter 88-20, Supplement 4, Duke completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter dated June 21, 1994 (Ref. 10). The report contained a summary of the methods, results and conclusions of the Catawba IPEEE program. The IPEEE process and supporting Catawba PRA included a comprehensive, systematic examination of severe accident potential resulting from external initiating events. By letter dated April 12, 1999 (Ref. 11), the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter (page 6) states:

"The staff finds the licensee's IPEEE submittal is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and the IPEEE results are reasonable given the Catawba design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Catawba IPEEE has met the intent of Supplement 4 to GL 88-20."

In 1996, Catawba initiated Revision 2 of the 1992 IPE and provided the results to the NRC in 1998 (Ref. 12). In April 2001, Duke notified the NRC (Ref. 13) that a voluntary initiative at the Catawba Nuclear Station to provide backup cooling to the 1A and 2A high head safety injection Centrifugal Charging (NV) Pumps had been completed. In conjunction with the completion of the plant modifications, the Catawba PRA Level 1 analysis was also updated and was designated as Revision 2b. The impact of this modification was to lower the base case core damage frequency (CDF).

Revision 3 of the Catawba PRA was completed in December 2004. This update was a comprehensive revision to the PRA models and associated documentation. The objectives of this update were as follows:

- To ensure the models comprising the PRA accurately reflect the current plant, including its physical configurations, operating procedures, maintenance practices, etc.
- To review recent operating experience with respect to updating the frequency of plant transients, failure rates, and maintenance unavailability data.
- To correct items identified as errors and implement PRA enhancements as needed.
- To address areas for improvement identified in the recent Catawba PRA Peer Review.
- To utilize updated Common Cause Analysis data and Human Reliability Analysis data.

PRA maintenance encompasses the identification and evaluation of new information into the PRA and typically involves minor modifications to the plant model. PRA maintenance and updates as well as guidance for developing PRA data and evaluation of plant modifications, are governed by workplace procedures.

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the PRA is ensured is by maintaining a set of system notebooks on each of the PRA systems. Each system PRA analyst is responsible for updating a specific system model. This update consists of a comprehensive review of the system including drawings and plant modifications made since the last update as well as implementation of any PRA change notices that may exist on the system. The analyst's primary focal point is with the system engineer at the site. The system engineer provides information for the update as needed. The analyst will review the PRA model with the system engineer and as necessary, conduct a system walkdown with the system engineer.

The system notebooks contain, but are not limited to, documentation on system design, testing and maintenance practices, success criteria, assumptions, descriptions of the reliability data, as well as the results of the quantification. The system notebooks are reviewed and signed off by a second independent person and are approved by the manager of the group.

When any change to the PRA is identified, the same three-signature process of identification, review, and approval is utilized to ensure that the change is valid and that it receives the proper priority.

In January 2001, an enhanced manual configuration control process was implemented to more effectively track, evaluate, and implement PRA changes to better ensure the PRA reflects the as-built, as-operated plant. This process was further enhanced in July 2002 with the implementation of an electronic PRA change tracking tool. All plant modifications and emergency procedure changes are reviewed for PRA impact. Any items having a PRA impact are logged into the tracking tool and a risk assessment is performed on them. Any plant modifications that are assessed as having a medium or high risk impact are reviewed for each PRA application being performed. If a plant modification is considered to be significant to the PRA, per workplace procedures, an interim PRA update may be performed.

Peer Review Process

Between March 18-22, 2002, Catawba participated in the Westinghouse Owners Group (WOG) PRA Certification Program. This review followed a process that was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the Nuclear Energy Institute Risk (NEI) Applications Task Force. The resulting industry document, NEI-00-02 (Ref. 14), describes the overall PRA peer review process. The Certification/Peer Review process is also linked to the ASME PRA Standard (Ref. 15).

(Note: NEI has developed draft guidance for self-assessments to address the use of industry peer review results in demonstrating conformance with the ASME PRA standard. This additional guidance is intended to be incorporated into a revision of NEI-00-02. NRC Regulatory Guide 1.200 For Trial Use (An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, February 2004) provides the staff position on the ASME standard. Catawba plans to conduct a self-assessment against the ASME standard in the future.)

The objective of the PRA Peer Review process is to provide a method for establishing the technical quality and adequacy of a PRA for a range of potential risk-informed plant applications for which the PRA may be used. The PRA Peer Review process employs a team of PRA and system analysts, who possess significant expertise in PRA development and PRA applications. The team uses checklists to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA being reviewed. One of the key parts of the review is an assessment of the maintenance and update process to ensure the PRA reflects the as-built plant.

The review team for the Catawba PRA Peer Review consisted of six members. Three of the members were PRA personnel from other utilities. The remaining three were industry consultants. Reviewer independence was maintained by assuring that none of the six individuals had any involvement in the development of the Catawba PRA or IPE. A summary of some of the Catawba PRA strengths and recommended areas for improvement from the peer review are as follows:

Strengths

- Aggressive response to past PRA peer reviews
- Knowledgeable personnel
- Culture of continuous improvement
- Documentation of final results and analyses
- Good capture of plant experience into the model
- Rigorous Level 2 and 3 PRA

Recommended Areas for Improvement

- Limited comparison to other plant / utilities PRAs for results and techniques
- Better documentation of bases for success criteria and HRA timing
- More focus on realism vs. conservatism in models
- More attention to eliminating old documentation and modeling assumptions / simplifications
- Consider more efficient methods to streamline recovery / post-processing process

The significance levels of the WOG Peer Review Certification process have the following definitions:

A. Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.
B. Important and necessary to address but may be deferred until the next PRA update.

Based on the PRA peer review report, the Catawba PRA received no Fact and Observations (F&O) with the significance level of "A" and 32 F&O with the significance level of "B". The "B" findings have been reviewed and prioritized for incorporation into the PRA. Thirteen of the "B" F&O have already been incorporated into Revision 3 of the PRA. It is expected that the remaining F&O will be resolved and incorporated into Revision 4 of the Catawba PRA.

The remaining open "B" F&O were reviewed with respect to any impact on the proposed LAR. It was determined that these have a negligible impact on the proposed LAR. A discussion of peer review items related to this LAR and their disposition is provided in Attachment 3.

<u>Results of Reviews with Respect to this LAR</u>

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 was documented in Duke Calculations. The review verified that the calculations adequately modeled the effects of the NSW system's 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 the original amendment request package.

3.3 Schedule Considerations for NSWS Outage

Presently it is estimated that the work required in taking the system out of service and draining the affected portions, will take approximately 1 day. The affected sections of piping will be cleaned which should take approximately 3 - 4 days. After cleaning, this evolution will include inspection and evaluation of the NSWS piping. The inspection results will be evaluated for repairs and/or coatings for the welds. After inspection, the welds in the affected piping will be coated and allowed to cure. This portion should take approximately 6 - 7 days. Upon completion, Operations will be required to fill the NSWS, and perform any necessary post maintenance testing which should take approximately 2 days. Therefore, the total time should run from 12 - 14 days. This project is being carefully scheduled to minimize the outage time. Catawba is requesting a one-time TS extension for up to 14 days (11 days beyond the current 72 hour AOT) for each NSWS train.

After careful consideration Catawba has determined to perform these activities when both units are in Mode 1, at 100% power. Catawba has performed NSWS pipe cleaning with one unit in a refueling outage and one unit at 100% power and performed NSWS pipe replacement with both units at 100% power. In both cases the work was completed safely and within the extended time granted by the NRC. There are different issues to manage with both schedules. During refueling outages, there are many important activities ongoing with many additional personnel located onsite. Several activities occur that affect power supply and distribution. In order to effectively implement the station's Defense-In-Depth risk management program for decay heat removal, it is advantageous and prudent to curtail any NSWS work during periods of high decay heat prior to core unloading as well as during periods following core reload when the refueling cavity is drained and the RCS loops are not filled. This would necessitate integrating this NSWS project with outage requirements and activities that can be mutually exclusive. In addition to scheduling difficulties, the chances for error and the possibility of plant events would be significantly increased. Furthermore, refueling outages are typically scheduled during the spring or fall when inclement weather is more likely to occur. This could adversely impact the NSWS system upgrades. Performing the work with both units at 100% power allows more flexibility in scheduling around inclement weather periods and allows for more focused management and site attention. Thus, the whole site can be focused on this project as opposed to several projects that are typically occurring during a refueling outage. This was clearly evident during the NSWS pipe replacement in January 2003. Therefore, after careful consideration of the above discussion, Catawba has decided to pursue performing the NSWS enhancements with both units at 100% power.

The scheduling of the NSWS header outages has taken into account the potential for severe weather. Refueling outages are typically scheduled during the spring (thunderstorms) or early fall (hurricane peak season) when severe weather is more likely to occur. This could adversely impact the NSWS system upgrades. Therefore, the NSWS outages will be scheduled for either the late fall or winter. This would be after the peak hurricane season and prior to the beginning of the spring weather which can produce frequent severe weather. In addition, the peak season for tornadoes tends to be in the spring and the peak season for thunderstorms tends to be in the summer. Since the incidence of severe weather would be greater than at other times of the year, the risk of a loss of offsite power (LOOP) during these time periods is also greater.

Catawba has developed contingency plans to react as needed to unforeseen weather changes. Catawba has a procedure, RP/O/B/5000/030, Severe Weather Preparations, which would be used in the event of severe weather. This procedure would be implemented by Emergency Planning with the concurrence of the Station Manager (designee) when sustained high speed winds greater than 50 mph are forecast for the site. The procedure provides guidance for various plant departments on prudent actions to be taken for approaching severe weather.

Catawba will monitor the National Weather Service reports prior to commencing and for the duration of each NSWS header outage. This will be done to ensure, to the extent practicable, that any potential outbreaks of severe weather are factored into the schedule and if severe weather should occur that appropriate personnel are notified and appropriate actions taken.

4.0 Technical Analysis

4.1 TS Systems Affected by NSWS Outage

An evaluation of the impact of these proposed temporary TS changes on other safety systems was performed. The effect of modified operation of the ECCS, CSS, CVIWS, AFW, CCW, NSWS, CRAVS, ABFVES, and EDG systems due to the NSWS activities on equipment required by other TS as well as effect of other TS during the one time 14-day period for each train was evaluated. The proposed temporary TS changes discussed in section 2.0 of this enclosure address the conclusions of this evaluation.

NSWS TS 3.7.8 only requires additional entry into TS 3.8.1 for the associated EDG and TS 3.4.6, "Reactor Coolant System Loops - Mode 4," for the associated RHR loop made inoperable by the inoperable NSWS train. During the pipe replacement project, both units will be in Mode 1, so the requirement to enter TS 3.4.6 will not be applicable. No other TS are required by TS 3.7.8 to be directly entered. Since the inoperability of NSWS results in the inoperability of the associated DG, TS that rely on DG operability will have to be entered.

The containment spray system relies on NSWS flow through containment spray system heat exchangers during the recirculation phase of a LOCA. Therefore, during the "A" & "B" NSWS loop outages, NSWS flow will be isolated to its respective containment spray system heat exchanger. In this condition the containment spray system train with its NSWS supply isolated will be considered inoperable. This results in entry into the TS LCO for TS 3.6.6 for containment spray system during the time in the project when a NSWS loop is inoperable.

NSWS is the safety related assured source for make up water supply to the CVIWS during a postulated accident. During each NSWS loop outage, NSWS flow will be isolated to its respective CVIWS train. In this condition the CVIWS train with its NSWS supply isolated will be considered inoperable. This results in entry into the TS LCO for TS 3.6.17 for CVIWS during the time in the project when a NSWS loop is inoperable.

During the "A" & "B" NSWS loop outages, NSWS flow will be isolated to its respective CCW heat exchanger. During this alignment, Operations will rack out the respective CCW pump motor breakers. Also the loads on the CCW trains will be in a cross tie alignment. In this condition the CCW train with its NSWS supply isolated will be considered inoperable. This results in entry into the TS LCO for TS 3.7.7 for CCW during the time in the project when a NSWS loop is inoperable.

Catawba operating procedures for CCW cross tie alignment are written to maintain availability of essential heat loads associated with the CCW train made unavailable when the CCW system is in a cross tie alignment except for the heat exchangers associated with the RHR and CCW trains.

The Residual Heat Removal Heat Exchanger associated with the inoperable CCW train would not be aligned to the operable CCW train. The RHR Heat Exchanger isolation valve associated with the inoperable train is secured by closing the valve and opening its breaker. This causes entry into TS 3.5.2, ECCS - Operating for both units during the time in the project when the NSWS loop is inoperable.

Other systems covered by TS are addressed by TS 3.0.6. TS 3.0.6 requires that when a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with TS 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists is required to be entered.

The AFW system is an exception to TS 3.0.6 because of the wording in the Bases section for the LCO. The NSWS is the safety-related source of water supply to the AFW system. During the "A" & "B" NSWS loop outages, this source will be taken out of service for up to 14 days. This will affect the safety related water supply to the AFW motor driven pumps that are aligned to the NSWS loop that is out of service. The opposite train motor driven AFW pumps and the turbine AFW pump on each unit will still have a safety-related source of water supply from the operable train of NSWS.

4.2 System Descriptions

Nuclear Service Water System

The NSWS provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident. During normal operation and during normal plant shutdowns, the NSWS also provides this function for various safety related and non-safety-related components.

The NSWS consists of two independent loops (designated A and B) of essential equipment, each of which is shared between the two units. Each loop contains two NSWS pumps, each of which is provided backup emergency power from a separate emergency diesel generator (EDG). 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 EDG is available. The NSWS also provides a safety-related source of water for the Auxiliary Feedwater (AFW) system.

Emergency Core Cooling System

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

Containment Spray System

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA).

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).

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

<u>Containment Valve Injection Water System</u>

The CVIWS is designed to inject water between the two seating surfaces of double disc gate valves used for Containment isolation. The injection pressure is higher than Containment design peak pressure during a LOCA. This will prevent leakage of the Containment atmosphere through the gate valves, thereby reducing potential offsite dose below the values specified by 10 CFR 100 limits following the postulated accident.

The system consists of two independent, redundant trains; one supplying gate valves that are powered by the A train diesel and the other supplying gate valves powered by the B train diesel. This separation of trains prevents the possibility of both containment isolation valves not sealing due to a single failure.

Each train consists of a surge chamber, which is filled with water and pressurized with nitrogen. One main header exits the chamber and splits into several headers. A solenoid valve is located in the main header before any of the branch headers, which will open after a 60 second, delay on a Phase A isolation signal. Each of the headers supply injection water to containment isolation valves located in the same general location, and close on the same engineered safety signal. A solenoid valve is located in each header, which supplies seal water to valves closing on a Containment Pressure - High-High signal. These solenoid valves open after a 60 second delay on a Containment Pressure - High-High signal. Since a Phase A isolation signal occurs before a Containment Pressure - High-High signal, the solenoid valve located in the main header will already be injecting water to Containment isolation valves closing on a Phase A isolation signal. This leaves an open path to the headers supplying injection water on a Containment Pressure - High-High signal. The delay for the solenoid valves opening is to allow adequate time for the slowest gate valve to close, before water is injected into the valve seat.

Makeup water is provided from the Demineralized Water Storage Tank for testing and adding water to the surge chamber during normal plant operation. Assured water is provided from the essential header of the NSWS. This supply is assured for at least 30 days following a postulated accident. If the water level in the surge chamber drops below the low-low level or if the surge chamber nitrogen pressure drops below the low-low pressure after a Phase A isolation signal, a solenoid valve in the supply line from the NSWS will automatically open and remains open, assuring makeup to the CVIWS at a pressure greater than 110% of peak Containment accident pressure.

Auxiliary Feedwater System

The AFW System is configured into three trains. The AFW System is considered operable when the components and flow paths required to provide redundant AFW flow to the steam generators are operable. This requires that the two motor driven AFW pumps be operable in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be operable with redundant steam supplies from two main steam lines upstream of the Main Steam Isolation Valves (MSIV), and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be operable. 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.

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. An additional source of supply is available from the Condenser Circulating Water System for safe shutdown events.

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 CSS. 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 Standby Nuclear Service Water Pond serves as the ultimate long-term safety related source of water for the AFW System. The automatic detection and transfer controls of the AFW System will detect and transfer the pump suctions to nuclear service water upon detection of the postulated failures of the condensate supplies.

Component Cooling Water System

The CCW System 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, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the NSWS, and thus to the environment. The CCW System is arranged as two independent, full capacity cooling loops, and has isolatable non-safety related components. Each safety related train includes two 50% capacity pumps, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus.

Control Room Area Ventilation System

The CRAVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity or high chlorine gas. The CRAVS consists of two independent, redundant trains that recirculate and filter the control room area air. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as prefilters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filters. The CRAVS is shared between the two units. The system must be operable for each unit when that unit is in the mode of applicability. Additionally, both normal and emergency power must also be operable because the system is shared. If a CRAVS component becomes inoperable, or normal or emergency power to a CRAVS component 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.

Auxiliary Building Filtered Ventilation Exhaust System

The ABFVES normally filters air exhausted from potentially contaminated areas of the auxiliary building, which includes the Emergency Core Cooling System (ECCS) area and non-safety portions of the auxiliary building. The ABFVES, in conjunction with other normally operating systems, also provides ventilation for these areas of the auxiliary building. The ABFVES consists of two independent and redundant trains. Each train consists of a heater demister section and a filter unit section. The heater demister section consists of a prefilter/moisture separator (to remove entrained water droplets and to prevent excessive loading of the carbon adsorber) and an electric heater (to reduce the relative humidity of air entering the filter unit). The filter unit section consists of a prefilter, an upstream HEPA filter, an activated carbon adsorber (for the removal of gaseous activity, principally iodines), a downstream HEPA, and a fan.

Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), the ABFVES exhausts air from the ECCS pump rooms while remaining portions of the system are isolated. This exhaust air goes through the pump room heater demister. The pump room heater demister removes both large particles within the air and entrained water droplets present in the air. The heater demister also preheats air and reduces the relative humidity of the air prior to entry into the filter unit. The pump room heater demister prevents excessive loading of the HEPA filters and carbon absorbers within the filter unit.

The ABFVES fans power supply is provided by electrical buses, which are shared between the two units. If normal or emergency power to the ABFVES 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.

Emergency Diesel Generators

Each train of the 4.16 kV Essential Auxiliary Power System is provided with a separate and independent emergency diesel generator (EDG) to supply the Class 1E loads required to safely shut down the unit following a design basis accident. Additionally, each EDG is capable of supplying its associated 4.16 kV blackout switchgear through a connection with the 4.16 kV essential switchgear.

Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions.

The Diesel Generator Engine Cooling Water System for each diesel includes a jacket water-intercooler water heat exchanger located within the Diesel Room, which is supplied with cooling water from the Nuclear Service Water System. The Diesel Generator Engine Cooling Water System is designed to maintain the temperature of the diesel generator engine within an optimum operating range during standby and during full-load operation in order to assure its fast starting and load-accepting capability and to reduce thermal stresses.

The system is also designed to supply cooling water to the engine lube oil cooler, the combustion air aftercoolers, and the governor lube oil cooler.

4.3 Defense in Depth

The proposed change to extend the AOT for each NSWS header and affected systems maintains the system redundancy, independence, and diversity commensurate with the expected challenges to system operation. The opposite train of emergency power and the associated engineered safety equipment remain operable during each NSWS train outage to mitigate the consequences of any previously analyzed accident.

In addition to the TS, the Work Control Program, Work Process Manual and associated procedures & programs that implement the Maintenance Rule (a)(4) Program provide for controls and assessments to preclude the possibility of simultaneous outages of redundant trains and to ensure system reliability. The proposed change to extend the AOT for each NSWS header will not alter the assumptions relative to the causes or mitigation of an accident.

The proposed change is required to meet the defense-in-depth principle consisting of a number of elements. These elements and the impact of the proposed change on each of these elements are as follows:

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

  The proposed Allowed Outage Time change does have an impact on CDF and large early release frequency (LERF). This impact is offset by the additional actions taken as described in the following bullet. Also, because this is a temporary and not a permanent change, the time averaged risk increase is acceptable. The increase in the overall reliability of the NSWS along with the decreased unavailability in the future because of this project will result in an overall increase in the safety of both Catawba units. Therefore, the change does not degrade core damage prevention and compensate with improved containment integrity nor do these changes degrade containment integrity and compensate with improved core damage prevention. The balance between prevention of core damage and prevention of containment failure is maintained. Consequence mitigation remains unaffected by the proposed changes. Furthermore, no new accident or transients are introduced with the requested change.

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

  Safety systems will still function in the same manner with the same reliability. The following compensatory measures will be taken to provide additional assurance that public health and safety will not be adversely affected by this request. These actions will be applied to both Units 1 and 2 as necessary unless otherwise specified.

  - During each 14-day period when operating with only one operable NSWS header, no major maintenance or testing will be planned on the remaining operable NSWS header. In addition, during each 14-day period, no major maintenance or testing will be planned on the operable equipment that relies upon NSWS as a support system. To the maximum extent practicable, routine tests (e.g. quarterly pump tests) and preventive maintenance work (e.g. motor checks) will be scheduled prior to or following each 14-day period. Certain tests may have to be performed during each 14-day period.

  - Diesel Generator Jacket Water Heat Exchanger - A temporary Engineering Change will be installed on each train of EDGs on both units to maintain the technically inoperable EDG capable of being manually started while the normal NSWS supply piping is out of service. This will be accomplished by using water from the fire protection system.

  - Diesel Generator Starting Air - An Engineering Change will be installed on each train of EDGs on both units to maintain the cooling water to the diesel generator starting air system aftercoolers while the normal NSWS supply piping is out of service. This will be accomplished by using drinking water to supply the aftercooler. This cooling water flow rate is adequate to maintain the non safety-related function of the starting air compressors.

  - No major maintenance or testing will be planned on the operable offsite power sources during each 14 day period. Switchyard activities will be coordinated to ensure that the operable offsite power supply and main transformer on both units are protected to the maximum extent practicable.

- Appropriate training will be provided to Operations personnel on this TS change, contingency measures to be implemented during each 14 day period, and actions to be taken in the event of flooding in the turbine building. Also, Operations will review the loss of NSWS and loss of CCW procedures as well as perform extra rounds on the CCW system.

- During each 14-day period, no major maintenance or testing will be planned on the Standby Shutdown Facility (SSF). To the maximum extent practicable, routine tests and preventive maintenance work for the SSF will be scheduled prior to or following each 14-day period.

- During each 14-day period, no major maintenance or testing will be planned on the operable trains of ECCS, CSS, CVIWS, AFW, CCW, CRAVS, ABFVES, and EDG. Routine tests and preventive maintenance work for these systems will be scheduled prior to or following each 14-day period. These items are being done to ensure the operable trains are protected to the maximum extent practicable.

- During each 14-day period that a NSWS header is out of service, the operable trains remaining in service will be considered protected trains. Operations will increase their routine monitoring of these trains to help ensure their operability. This increase in routine monitoring will also include the Turbine Building to ensure no flooding in this area.

- Plant procedures will be used to cross tie selected CCW system loads during the time period a CCW heat exchanger will be out of service during the NSWS pipe replacement.

- The Turbine Building flooding comprises several of the accident sequences. Catawba has installed permanent flood protection barriers in the turbine building to mitigate this issue. Operators will also review actions to be taken in the event of flooding in the Turbine Building. Both of these actions decrease the time to react to internal flooding transients and therefore result in a reduction of risk.

- 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 NSWS pipe replacement project.

- The PRA model has been revised since the first submittal on November 16, 2004. The revised model was used for a new PRA calculation. This calculation identified a scenario that could be remediated with some operator action. As a result, a procedure change(s) will be made to ensure that an operator is stationed at the correct time to control the Unit 1 and Unit 2 auxiliary feedwater flow control valves in the event that flow control is lost following a loss of 4160V ac power on Unit 1 or Unit 2 as applicable. One of the more important operator actions as identified in the PRA is manually throttling the auxiliary feedwater flow to the steam generators following a loss of 4160V ac power. These procedure changes will ensure that the importance of this action is communicated to Operations and the improved operator response to these events results in a reduction of risk over that identified in the PRA.

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

  There is no impact on the redundancy, independence, or diversity of the systems described in this TS amendment or on the ability of the plant to respond to events with diverse systems. The systems described in this TS amendment are diverse and redundant systems and will remain so. The following discussion addresses each system affected by this TS change.

  NSWS
  During this time period the operable NSWS loop will be protected to the extent practical by minimizing any

maintenance on the system for either unit. In this configuration, the operable loop will still respond as designed during design basis events.

ECCS
During the time when a NSWS loop is out of service, the respective ECCS equipment on the CCW train without NSWS cooling will be supplied from the opposite CCW train via a cross train alignment. In this cross train alignment selected essential heat loads, except for the heat exchangers associated with the RHR and CCW systems, for the CCW train made inoperable will be supplied by the operable CCW train.

During this time period the operable ECCS train will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

CS
During this time period the operable containment spray system train will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

CVIWS
During the NSWS system upgrades this assured makeup flow would not be available during the time frame that each NSWS loop is out of service. However this does not affect the operation of the system during the initial phase of a postulated accident.

During this time period the operable CVIWS train will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

AFW
During this time period the operable AFW trains will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable trains will still respond as designed during design basis events.

CCW
During this time period the operable CCW train will be protected to the extent practical by minimizing any

maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. In this Condition, the remaining operable CCW train is adequate to perform the heat removal function.

CRAVS

During this time period the operable CRAVS train will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

ABFVES

During this time period the operable ABFVES train will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

EDGs

During the NSWS project, the NSWS supply to one EDG on each unit will be inoperable. An Engineering Change will be implemented for the EDGs on each unit without their NSWS supply to supply an alternate, non-safety related, source of cooling to the EDG with the inoperable NSWS supply. The EDG will be considered inoperable, but it will be technically capable of being manually started to perform its intended function.

During this time period the operable EDG will be protected to the extent practical by minimizing any maintenance on the system for either unit. In this configuration, the operable train will still respond as designed during design basis events.

- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.

Defenses against common cause failures are maintained. The extended AOT for each NSWS train requested is not sufficiently long to expect new common cause failure mechanisms to arise. In addition, the operating environment for these components remains the same so; again, new common cause failures modes are not expected. In addition, backup systems are not impacted by this change and no new common cause links between the primary and backup systems are introduced. Therefore, no new potential common cause failure mechanisms have been introduced by the proposed change.

- Independence of barriers is not degraded.

  The barriers protecting the public and the independence of these barriers are maintained. Multiple systems will not be taken out of service simultaneously that could lead to degradation of these barriers and an increase in risk to the public. During each NSWS header outage, the redundant train of equipment will remain operable and capable of performing its intended function. In addition, the extended AOT for each NSWS train does not provide a mechanism that degrades the independence of the barriers, fuel cladding, reactor coolant system, and containment.

- Defenses against human errors are maintained.

  Administrative controls have been implemented to reflect the contingency measures that are being established. The increase in the AOT for each NSWS header outage will provide the necessary time to implement system upgrades which include activities associated with cleaning, inspection, and coating of NSWS piping welds, and necessary system repairs, replacement, or modifications. This will reduce time pressure during this project which will facilitate improved operator and maintenance personnel performance resulting in reduced system alignments and assembly errors.

  Performing the work with both units at 100% power allows more flexibility in scheduling around inclement weather periods and allows for more focused management and site attention. Thus, the whole site can be focused on this project as opposed to several projects that are typically occurring during a refueling outage.

It is concluded that defense-in-depth was not impacted by the proposed changes.

4.4 Safety Margin Assessment

The overall margin of safety is not decreased due to the extended AOT for each NSWS header since the system design and operation are not altered by the proposed extended AOT. As described in the section 4.3, Defense in Depth, the design, operation, and response of the systems addressed in this extended AOT are not affected. They will continue to operate as designed and the plant has controls in place to prevent removing redundant trains of equipment at the same time.

The safety analysis acceptance criteria stated in the Updated Final Safety Analysis Report (UFSAR) are not impacted by the change. Redundancy and diversity of the AFW System will be maintained. The proposed change will not allow plant operation in a configuration outside the design basis of the plant. The system requirements credited in the accident analysis will remain the same. It was concluded that safety margins were not impacted by the proposed changes.

4.5 Additional Plant Systems

A separate plant system has been incorporated into the Catawba design to allow a means of limited plant shutdown, independent from the control room and auxiliary shutdown panels. This system, known as the Standby Shutdown System, provides an alternate means to achieve and maintain a hot shutdown condition following postulated fire and sabotage events. This system is in addition to the normal shutdown capabilities available. The Standby Shutdown System (except for interfaces to existing safety-related systems) is designed in accordance with accepted fire protection and security requirements and is not designed as a safety related system. The Standby Shutdown System utilizes the turbine driven AFW pump to provide adequate secondary side makeup independent from all AC power and normal sources of water. During this mode of operation, the turbine driven AFW pump operates remotely controlled from the Standby Shutdown Facility (SSF). If the turbine has not started automatically prior to the event, it may be started manually and receive suction water from condensate sources. If condensate sources are depleted or lost, the turbine will automatically transfer suction to an independent source initiated by the SSF related train of the condensate source loss detection logic and battery-powered motor-operated valves. The independent source of water is the buried piping of the Condenser Circulating Water System, which contains sufficient water in the embedded pipe to maintain the plant at hot standby for at least 3 days. In this manner, sufficient AFW flow may be maintained even if all normal and emergency AC power is lost, and all condensate and safety-grade water sources are lost.

In order to improve the total core damage frequency, backup cooling was provided to Centrifugal Charging Pump (CCP) 1A (2A). The backup cooling water to CCP 1A (2A) is supplied by a non-safety related four-inch drinking water system header in the Auxiliary Building. The drinking water system supply ties into the CCW System Supply piping to the CCP 1A (2A) Motor Coolers and Pump Bearing and Speed Reducer Oil Coolers. On the CCW System return side of these coolers, drain lines are routed from the return lines to the containment spray/residual heat removal sump in the Auxiliary Building. The backup cooling water can be aligned to either the 1A or 2A CCP but not to both pumps at the same time. The backup cooling supplied by the drinking water system is not safety-related and is not relied upon to mitigate any design basis accidents or events. Operability of the "A" CCP is not dependent on the backup cooling.

4.6 Evaluation of Risk Impact

Duke Power has used a risk-informed approach to determine the risk significance of taking a loop of NSWS out of service for up to 11 days beyond its current TS limit of 72 hours. The acceptance guidelines given in the EPRI PSA Applications Guide were used as a gauge to determine the significance of the short-term risk increase from the outage extension.

The current PRA model was used to perform the risk evaluation for taking a train of NSWS out of service beyond its TS limit. The quantification has taken into account the following specific conditions:

- No discretionary maintenance is planned for risk significant components such as AFW, RHR, NV, CCW, 4160V ac power, DGs, and the SSF.
- Historically, the major source of flooding in the Turbine Building has been the Condenser Circulating Water (RC) System. Flood walls have recently been installed in the Turbine Building to protect the station transformers that feed the 4160 volt busses (SATA, SATB, 1(2) ATC, and 1(2) ATD). In the current model, the flood initiator as currently defined would be eliminated; however, some new less severe flooding events may be created. The impact of turbine building

flooding given the new walls has been included in the assessment.

The estimated increase in the core damage probability for Catawba for each NSWS loop outage ranges from 2.7E-06 for a 2-day extension up to 1.5E-05 for an 11-day extension.

The impact to the seismic core damage frequency (CDF) was also evaluated. The NSWS components and piping have been assessed to be seismically-rugged and the electrical systems become the dominant failure mechanism. Given that the EDGs and switchyard will be available during the NSWS upgrades, there are no new failure modes introduced. The increase in core damage probability for the CT from the seismic initiator is negligible compared to the non-seismic contribution.

It is also recognized that reductions in risk can be achieved by the consideration of several other non-quantifiable risk reduction factors:

- Based upon a review of the cut sets, a substantial portion of the accident sequences involve a loss of 4160V ac power, NSWS or CCW. Therefore, it would be prudent for the operators to review the loss of power, loss of NSWS and loss of CCW procedures as well as perform extra rounds on the CCW System.
- No maintenance or testing should be performed on the offsite power system (switchyard). The operability of required offsite circuits should also be maintained. These actions would reduce the likelihood of losing off site power and therefore reduce risk.
- Note that the peak season for tornadoes tends to be in the spring and the peak season for thunderstorms tends to be in the summer. Since the incidence of severe weather would be greater than at other times of the year, the risk of a LOOP during these time periods is also greater. (The PRA uses a yearly average initiating event frequency.) The NSWS piping AOTs are scheduled to be performed during a time of the year when severe weather is not normally an issue; hence, the risk of a severe weather-related LOOP is minimized. (It is noted, however, that the site has developed contingency plans to react as needed to unforeseen weather changes.)
- Entry into and operation of shutdown cooling is not without risk and involves significant plant manipulations and evolutions on both the primary and

secondary side by Operations personnel. This risk is averted by remaining at power.

The change in the conditional large early release probability (CLERP) was also evaluated. The estimated increase in the large early release probability for Catawba during the NSWS loop outage ranges from 5.4E-08 for a 2-day extension up to 3.4E-07 for an 11-day extension. It is concluded that the large early release probability (LERP) implications of the extended LCO are not significant.

The core damage frequency contribution from the proposed outage extension is judged to be acceptable for a one-time, or rare, evolution. As stated above, the estimated increase in the core damage probability for Catawba for each NSWS loop outage ranges from 2.7E-06 for a 2-day extension up to 1.5E-05 for an 11-day extension.

4.7 Summary

The "A" & "B" train NSWS pipe cleaning and weld coating project and the proposed temporary changes to TS 3.5.2, 3.6.6, 3.6.17, 3.7.5, 3.7.7, 3.7.8, 3.7.10, 3.7.12, and 3.8.1 have been evaluated to assess their impact on the normal operation of the affected systems and to ensure that the design basis of these functions are preserved.

The requested period of 14 days (336 hours) for completing the Required Action is reasonable considering the redundant capabilities of the system, the proposed contingency measures that will be taken as discussed in section 4.3 of this Enclosure, the additional plant systems discussed in section 4.5 of this Enclosure, and the risk considerations discussed in section 4.6 of this Enclosure. Therefore, for the ECCS, CSS, AFW, CCW, NSWS, and EDG systems, the requested extension of the Required Action time from 72 hours to 336 hours is acceptable.

Catawba has requested an NSWS train outage extension of 11 days beyond the 72 hour TS allowance for *each train*. Per the most recent revision of the Catawba PRA (Revision 3), the baseline Core Damage Frequency (CDF) is 3.68E-05 / yr. (at a truncation limit of 5E-10) and the Large Early Release Frequency (LERF) is 2.70E-06 / yr. (at a truncation limit of 5E-11). For an 11-day period, the baseline Core Damage Probability (CDP) becomes 1.23E-06 and the Large Early Release Probability (LERP) is 9.04E-08. The licensee's analysis indicates that for an 11-day period when a train of NSWS is out of service, the Conditional Core Damage

Probability (CCDP) becomes 1.63E-05 and the Conditional Large Early Release Probability (CLERP) is 4.31E-07. Therefore, for each 11 day extension, the Incremental Core Damage Probability (ICDP) is 1.51E-05 and the Incremental Large Early Release Probability (ILERP) is 3.41E-07.

The requested period of 14 days (336 hours) for completing the Required Action is reasonable considering the redundant capabilities of the system, the proposed contingency measures that will be taken as discussed in section 4.3 of this Enclosure, the additional plant systems discussed in section 4.5 of this Enclosure, and the risk considerations discussed in section 4.6 of this Enclosure. Therefore, for the CVIWS, CRAVS, and ABFVES systems, the requested extension of the Required Action time from 168 hours to 336 hours is acceptable.

4.8 Precedent Licensing Actions

This proposed license amendment was modeled after two (2) similar license amendments previously granted by the NRC. The first amendment was granted for the Catawba Nuclear Station in support of the NSWS system upgrades. The NRC granted the license amendment in a SER for Amendments Nos. 189 and 182 on October 4, 2000. The second amendment was granted for the Catawba Nuclear Station in support of the replacement of a portion of the NSWS piping. The NRC granted the license amendment in a SER for Amendments Nos. 203 and 196 on January 7, 2003.

5.0 REGULATORY ANALYSIS

This section addresses the standards of 10 CFR 50.92 as well as the applicable regulatory requirements and acceptance criteria.

5.1 NO SIGNIFICANCE HAZARDS CONSIDERATIONS (NSHC)

Catawba is currently pursuing a project to repair a portion of both trains of the nuclear service water system (NSWS) piping for both units. This is necessary to maintain the long-term reliability of the NSWS. This project represents a challenge in that it is not possible to isolate, drain, replace, restore and test the NSWS during the current TS action time frame. The purpose of this submittal is to request a temporary change to the existing TS for the systems affected during the project. This will permit an orderly and efficient project implementation during power operation on both units. The specific change is to extend the TS required action time from 72 hours to 336 hours.

The increase in core damage frequency for the duration of the time period of single header operation during the supply header weld coating work is greater than 1E-6 (per PRA analysis). Although this increase in frequency is above the normally acceptable value (for a permanent change), the short term risk increase is offset by an increase in long term safety system reliability due to the improved supply header condition and NSWS system upgrades.

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

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

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The pipe repair project for the NSWS and proposed TS changes have been evaluated to assess their impact on normal operation of the systems affected and to ensure that the design basis safety functions are preserved. During the pipe repair the other NSWS train will be operable and no major maintenance or testing will be done on the operable train. The operable train will be protected to help ensure it would be available if called upon.

This pipe repair project will enhance the long term structural integrity in the NSWS system. This will ensure that the NSWS headers maintain their integrity to ensure its ability to comply with design basis requirements and increase the overall reliability for many years.

The increased NSWS train unavailability as a result of the implementation of this amendment does involve a one time increase in the probability or consequences of an accident previously evaluated during the time frame the NSWS headers are out of service for pipe repair. Considering this small time frame for the NSWS train outages with the increased reliability and the decrease in unavailability of the NSWS system in the future because of this project, the overall probability or consequences of an accident previously evaluated will decrease.

Therefore, because this is a temporary and not a permanent change, the time averaged risk increase is acceptable. The increase in the overall reliability of the NSWS along with the decreased unavailability in the future because of the pipe repair project will result in an overall increase in the safety of both Catawba units. Therefore, the consequences of an accident previously evaluated remains unaffected and there will be minimal impact on any accident consequences.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or *different kind of accident from any accident previously evaluated?*

Response: No.

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed temporary TS changes do not affect the basic operation of the ECCS, CSS, CVIWS, NSWS, AFW, CCW, CRAVS, ABFVES, or EDG systems. The only change is increasing the required action time frame from 72 hours (ECCS, CSS, NSWS, AFW, CCW, and EDG) or 168 hours (CVIWS, CRAVS and ABFVES) to 336 hours. The train not undergoing maintenance will be operable and capable of meeting its design requirements. Therefore, only the redundancy of the above systems is affected by the extension of the required action to 336 hours. During the project, contingency measures will be in place to provide additional assurance that the affected systems will be able to complete their design functions.

No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the plant, which will introduce any new accident causal mechanisms.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

Implementation of this amendment would not involve a significant reduction in a margin of safety. 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 impacted by implementation of this proposed temporary TS amendment. During the NSWS train outages, the affected systems will still be capable of performing their required functions and contingency measures will be in place to provide additional assurance that the affected systems will be maintained in a condition to be able to complete their design functions. No safety margins will be impacted.

The probabilistic risk analysis conducted for this proposed amendment demonstrated that the CDP associated with the outage extension is judged to be acceptable for a one-time or rare evolution. Therefore, there is not a significant reduction in the margin of safety.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment for a temporary one time TS change does not involve a significant hazards consideration.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The regulatory bases and guidance documents associated with the systems discussed in this proposed TS amendment include:

GDC-2 requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of capability to perform their safety functions.

GDC-4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

GDC-34 requires a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC-35 requires a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC-38 requires a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC-44 requires a system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

There will be no changes to the ECCS, CSS, CVIWS, NSWS, AFW, CCW, CRAVS, ABFVES, and EDG design such that compliance with any of the regulatory requirements and guidance documents above would come into question. The evaluations discussed in this enclosure confirm that the plant will continue to comply with applicable regulatory requirements.

The requested period of 14 days (336 hours) for completing the Required Action for the above systems is reasonable considering the redundant capabilities of the systems, the proposed contingency measures that will be taken as discussed in section 4.3 of this Enclosure, the additional plant systems discussed in section 4.5 of this Enclosure, and the risk considerations discussed in section 4.6 of this Enclosure. Therefore, the requested extension of the Required Action time to 336 hours is acceptable.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of this amendment will have no adverse impact upon the Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Catawba TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7.0 REFERENCES

1. Duke Letter from G. R. Peterson to U.S. Nuclear Regulatory Commission dated May 25, 2000.
2. Letter from C. P. Patel, Project Manager - U.S. Regulatory Commission to G. R. Peterson dated October 4, 2000, Issuance of License Amendments 189 and 182 for Catawba Nuclear Station.
3. Duke Letter from G. R. Peterson to U.S. Nuclear Regulatory Commission dated September 12, 2002.
4. Letter from R. E. Martin, Project Manager - U.S. Regulatory Commission to G. R. Peterson dated January 7, 2003, Issuance of License Amendments 203 and 196 for Catawba Nuclear Station.
5. "Catawba Nuclear Station Unit 1 Probabilistic Risk Assessment," Volumes 1-3, Duke Power Company, August 18, 1987.
6. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.
7. Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities, USNRC, November 1988
8. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "Individual Plant Examination (IPE) Submittal in Response to Generic Letter 88-20," September 10, 1992
9. Letter USNRC to Duke Power Company, "Safety Evaluation of Catawba Nuclear Station, Units 1 and 2 Individual Plant Examination (IPE) Submittal," June 7, 1994
10. Letter Duke Power Company to Document Control Desk (USNRC), Catawba Units 1 and 2, "Individual Plant Examination of External Events (IPEEE) Submittal," June 21, 1994.
11. Letter USNRC to Duke Power Company, "CATAWBA NUCLEAR STATION -- REVIEW OF INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)," April 12, 1999.
12. Letter Duke Energy Corporation to Document Control Desk (USNRC), Catawba Units 1 and 2, "Probabilistic Risk Assessment (PRA), Revision 2 Summary Report, January 1998."

13. Letter Duke Energy Corporation to Document Control Desk (USNRC), Catawba Units 1 and 2, "Centrifugal Charging Pump Modifications and Catawba PRA Update (Revision 2b)," April 18, 2001.

14. NEI-00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guideline," Nuclear Energy Institute, Revision A3, March 20, 2000.

15. "Standard For Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002 and addenda ASME RA-SC-2003, December 5, 2003.

16. Code of Federal Regulations 10CFR50.65; "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

17. NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.

18. NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 2000.

ATTACHMENT 1

ANSWERS TO NRC REQUESTS IN 2/16/05 LETTER

Answers to NRC Requests in February 16, 2005 Letter

NRC Requests:

The application should have further discussion on why the NSW piping is seismically qualified given that it is in a "degraded" condition. It also should present the specific actions and plans for severe weather mitigation.

Additionally, to better support the NRC's review of the risk impact of the extended NSWS AOT, additional information identified in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," should be provided. The areas of these RGs that should be addressed include:

1. Traditional Engineering Considerations (i.e. Defense in Depth);

2. Baseline Core Damage Frequency, Large Early Release Frequency, Conditional Core Damage Probability (CCDP), and Conditional Large Early Release Probability (CLERP) for the outage configurations (ensure that the calculated risk is clearly identifiable as incurred for each train outage, or as the combined risk impact of both outages);

3. Probabilistic risk assessment (PRA) scope and quality information, including a basis for the PRA model being representative of the as built, as operated plant, unit-specific differences, outstanding plant modifications or performance data not yet incorporated into the PRA models, model truncation levels, and key assumptions and uncertainties relevant to the amendment request;

4. Key plant and/or PRA model changes that have caused the risk calculations (i.e., CCDP) to change compared with the prior amendment requests;

5. Capability of the configuration risk management program;

6. How the additional unavailability incurred during extended NSW outages has been, and will be, addressed by Title 10 of the *Code of Federal Regulations* (10

CFR) Part 50.65 monitoring and performance criteria; and

7. Additional details and sensitivity analyses for specific PRA model features and assumptions identified in the submittal as important to maintaining overall risk during NSWS extended outages.

Catawba Response to NRC Requests:

- A discussion on seismic qualification is located in Section 3.0 on pages 1-7 and 1-8.
- A discussion on severe weather plans is located in Section 3.3 on page 1-21.

The following is the answers to the information requested from the referenced regulatory guides.

1. Section 4.0 was reformatted and additional information added to address this request.

2. The licensee has requested an NSWS train outage extension of 11 days beyond the 72 hr. TS allowance for each train. Per the most recent revision of the Catawba PRA (Revision 3), the average non-seismic baseline Core Damage Frequency (CDF) is 3.68E-05 / yr. and the Large Early Release Frequency (LERF) is 2.70E-06 / yr. For an 11-day period, the baseline Core Damage Probability (CDP) becomes 1.23E-06 and the Large Early Release Probability (LERP) is 9.04E-08 (assuming a capacity factor of 0.9). The licensee's analysis indicates that for an 11-day period when a train of NSWS is out of service, the non-seismic Conditional Core Damage Probability (CCDP) becomes 1.63E-05 and the Conditional Large Early Release Probability (CLERP) is 4.31E-07. Therefore, for each 11 day extension, the Incremental Conditional Core Damage Probability (ICCDP) is 1.51E-05 and the Incremental Conditional Large Early Release Probability (ICLERP) is 3.41E-07.

   The licensee's analysis also addressed the seismic contribution to the CDP. The increase in seismic CDP for the CT is negligible compared to the non-seismic contribution and was therefore considered to be low risk.

3. Section 3.2.3 was added to address this request.

4. Risk information for recent Catawba NSW LARs utilized Catawba PRA Revision 2c. This revision incorporated several significant changes into the PRA. These include the following:
   - Changes to the Auxiliary Feedwater Turbine-Driven Pump logic
   - Additional operator actions credited for small and medium LOCAs
   - Incorporation of new Reactor Coolant Pump seal package design
   - Addition of human error for failure to throttle Auxiliary Feedwater outside of the Control Room
   - Update to the Diesel / Generator unavailability data

   The current LAR contains risk information using Catawba PRA Revision 3. This revision also included several significant changes into the PRA. These include:
   - Improvements in breaking circular logic
   - Incorporation of new WOG ATWS modeling
   - System modeling updates
   - Human error dependency analysis

   During preparation of the current LAR, installation began on flood walls in the Turbine Building to protect transformers SATA, SATB, 1(2) ATC, and 1(2) ATD. The impact on the Turbine Building flood analysis is still undergoing evaluation and quantification. Logic changes will also be added to the AC power model. The analysis to support the current LAR includes a simplified treatment of the impact of the flood wall. The PRA will be revised and reissued for use upon the review and approval of the necessary changes regarding the flood wall installation.

5. Section 3.2.1 and 3.2.2 were revised to address this request.

6. Section 3.2.1 and 3.2.2 were revised to address this request.

7. Several important assumptions were made in the performance of the supporting LAR analysis. These include the following:
   - For certain accident sequences involving a loss of 4160V ac, it will be necessary to throttle Auxiliary Feedwater outside of the control room to prevent overfilling of the SGs. An increase in operator awareness in the control room as well as assigning an operator to be 'on call' expressly to perform the throttling function will reduce the time to diagnose and respond to this event, thus reducing the human error probability for this event.
   - Maintenance will not be performed on other key safety significant systems such as Auxiliary Feedwater, Residual Heat Removal, Chemical and Volume Control, 4160V ac power, the Diesel Generators, and the Safe Shutdown Facility.
   - The Diesel Generators (DGs) on the NSWS header undergoing maintenance will be available and the opposite train not undergoing maintenance will be fully operable on both units. Engineering Changes will be installed to provide cooling water to the DG jacket water coolers and DG starting air aftercoolers.

Accident sequences analyzed in the PRA model require action to throttle AFW outside of the control room to prevent overfilling of the SGs. The human error probability (HEP) quantification for this event is based upon sufficient times for the operators to diagnose the event and to successfully perform the throttling function. As a way of reducing the HEP during the extended completion time, the times to diagnose and respond are significantly reduced if operator awareness for these types of events is increased and an operator is 'on call' expressly to perform the throttling function. With these reduced times to diagnose and respond, the PRA analysis to support this LAR indicates a reduction in the CCDP for the 11-day completion time of approximately 50%. Therefore, the risk results for this application are very sensitive to this HEP.

Flood walls have recently been installed in the Turbine Building to protect station transformers that provide power to the 4160 volt busses (SATA, SATB, 1(2) ATC, and 1(2) ATD). With the installation of these walls, an internal flood occurring on one unit is not expected to affect the other. The supporting PRA analysis for this LAR estimated the impact to the Turbine Building flood initiator frequency. No appreciable decrease in the CCDP was shown when taking credit for the flood walls; therefore, the risk results for this application are insensitive to this installation.

Maintaining the key safety systems available during the NSWS completion times will minimize the risk associated with these activities by providing a level of redundancy to the systems and equipment used to mitigate a core damage accident. The relative importance of these mitigating systems was obtained by reviewing the cut set results from the 11-day extended completion time analysis. This serves as an 'independent means' of verifying the key safety systems. This process indicated the following systems were important to safety for this application: 4160V AC power, Auxiliary Feedwater, Diesels, Residual Heat Removal, Safe Shutdown Facility, Chemical and Volume Control, Safety Injection, and Vital dc power.

ATTACHMENT 2

SUMMARY OF REGULATORY COMMITMENTS

SUMMARY OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Duke Energy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Randall D. Hart, Regulatory Compliance, Catawba Nuclear Station (803) 831-3622.

COMMITMENT	Due Date/Event
The proposed changes to the Catawba Nuclear Station TS will be implemented within 60 days of NRC approval.	Within 60 days of NRC approval.
The contingency items discussed in section 4.3 of Enclosure 1 will be implemented during the extended allowed outage times for both the 'A' and 'B' NSWS train outages.	Prior to commencing the associated NSWS train outage.
Catawba will monitor the National Weather Service reports to ensure, to the extent practicable, that any potential outbreaks of severe weather are factored into the schedule and if severe weather should occur that appropriate personnel are notified and appropriate actions taken.	Prior to commencing and for the duration of each NSWS header outage.

ATTACHMENT 3

APPLICABLE PEER REVIEW ITEMS

Applicable Peer Review Items

Fact and Observation - Catawba	Resolution - Catawba
Determination of exposure times for Common Cause Factors involving a continuously running NSWS system is questioned.	The Catawba analysis uses an exposure time based on an estimated mean time to repair. While other approaches are possible, there is not currently a consensus among the industry on recommended exposure times for common-cause failure events associated with initiators. This is an industry generic issue that is being addressed. It is expected to have a negligible impact on the proposed LAR.
Exclusion of a potential diversion path for the Component Cooling System is questioned.	The risk impact of this change is expected to be low. In order for a diversion path to starve flow to required components, there would have to be failures of multiple components (pumps and valves). This has a negligible impact on the proposed LAR.
More recent generic data sources should be pursued.	Use of more recent equipment operating experience data is desirable. However, the current generic data is deemed to provide adequate estimates of generic equipment failure rates. In addition, these failure rates are Bayesian-updated with plant-specific data. This has a negligible impact on the proposed LAR.
The PRA flooding analysis does not address the effects of water on equipment.	Spray effects and flood propagation past flood barriers are not expected to be significant interactions. This is a more of a completeness / documentation issue than a actual issue since the general design of the plant considered spray effects. It was previously shown that the analysis was insensitive to Turbine Building flooding issues. It is expected that this has a negligible impact on the proposed LAR.
The initiating event frequencies for certain support system failures (NSWS, CCW) are not input in the top event logic as a Boolean equation, but rather as a point estimate whose value is derived by solution of the IE fault tree	Modeling initiators with logic instead of point estimates would improve the accuracy of the model, although it is not expected to significantly alter the results or conclusions of the analyses. Modeling initiators as point estimates is an accepted industry practice and is also used in the NRC SPAR models. Therefore, this has a negligible impact on the proposed LAR.